CONFERENCES AND SYMPOSIA

FUSION REACTOR MATERIALS

Report on the Second American Nuclear Society Topical Meeting,
Seattle, Washington, USA,
9–12 August 1981

N.M. GHONIEM, R.W. CONN
University of California,
School of Engineering and Applied Sciences,
Los Angeles, California,
United States of America

This international meeting on fusion reactor materials, consisting of a plenary invited session and parallel sessions of contributed and invited papers, was co-sponsored by the American Nuclear Society, the American Institute of Mining, Metallurgical and Petroleum Engineers, the American Society of Metals, the Office of Fusion Energy of the US Department of Energy, and the Electric Power Research Institute. The Technical Program Chairman was Dr. J.J. Holmes of the Hanford Engineering Development Laboratory (HEDL) and the Programme Committee consisted of representatives from the USA, the UK, the Federal Republic of Germany and Japan. The conference proceedings will be published in the Journal of Nuclear Materials. Consequently, the present report is intended to highlight the results of the conference.

1. OVERVIEW

The first international meeting on fusion reactor materials was held in 1979 in Miami Beach, Florida. The objective of the present conference was to review progress in materials research for fusion reactor applications during the past two years. Over 250 invited and contributed papers were presented. Fusion reactor materials research has shown considerable progress, with papers devoted to the following areas: (1) constraints in materials selection; (2) fundamental studies; (3) structural materials; (4) plasma/materials interactions; (5) compatibility between liquid metals and structures; (6) solid breeder materials; (7) special-purpose materials (insulators, ceramics, etc.); and (8) test methods.

The keynote address of the conference was given by C. Blattner, Vice President of Engineering, McDonnell-Douglas Astronautics Company. The thrust of his talk was that there are lessons to be learned by the fusion programme from the experience gained in the aerospace industry. Non-metallic and high-temperature alloys were particularly developed for the needs of this industry. Fusion energy will likewise require the development of materials especially tailored to the fusion environment. T. Reuther of the US Department of Energy reviewed the status and structure of the US fusion materials development.

The United States fusion materials programme is structured in five different tasks: (1) Alloy Development (ADIP); (2) Damage Analysis and Fundamental Studies (DAFS); (3) Plasma Materials Interactions (PMI); (4) Special-Purpose Materials (SPM); and (5) Analysis and Evaluation (AE). Within ADIP, structural material development is achieved through the following paths: (Path A) Austenitic 316 SS; (Path B) Ferritic Alloys; (Path C) Fe-Cr-Ni Alloys; (Path D) Refractory Metals; and (Path E) Innovative Materials such as long-range-ordered alloys or rapidly solidified alloys.

In order to achieve the objectives of the fusion materials programme, a variety of test facilities is used. In the USA, materials test facilities can generally be classified into:

1. Simulation sources: Mainly ion accelerators and high-voltage electron microscopes. Double and triple beam lines are used to study the fundamental interactions between displacement damage, helium, and hydrogen.

2. Fission facilities: These are mainly HFIR and ORR reactors (mixed neutron spectrum), and EBR-II and FFTF reactors (fast-neutron spectrum). One important parameter for radiation damage development is the ratio of helium atoms to displacement damage.

While HFIR produces too high a value for nickel-containing alloys, this ratio is too small in EBR-II, as compared to the real fusion conditions.

3. Fusion facilities: The main existing 14-MeV neutron facility is the RTNS-II at Lawrence Livermore Laboratory (LLL). The maximum attainable fluence is small (5×10^{18} n·cm^{-2}), and the test volume is not large. A major test facility for future irradiations with fusion neutrons is the FMIT facility to be constructed at HEDL. The current FMIT design indicates that the prime test volume having an average flux of 10^{15} n·cm^{-2}·s^{-1} is 21 ± 4 cm^3. There are 120 cm^3 in the test cell having a dpa rate greater than that experienced by a 1-MW·m^{-2} first wall.

The Japanese and European programmes on fusion materials have progressed considerably during the last few years. The status of fusion materials research in Japan was outlined by R.R. Hasiguti (University of Tokyo). The Japanese fusion materials programme is concentrated on materials development for tokamak-type reactors, and geared toward the JAERI large tokamak (JT-60) and the Experimental Fusion Reactor (EFR). Research on materials for fusion applications is contributed by:

(1) Japan Atomic Energy Research Establishment (JAERI)
(2) National Research Institute for Metals (NRIM)
(3) Universities.

The Japanese alloy development programme is structured in a fashion similar to the US programme, and has the following categories: (A) austenitic stainless steel, with the prime candidate being Ti-modified; (B) Fe-Cr-Ni alloys; (C) ferritic alloys; (D) refractory alloys; and (E) non-metallic materials. While the present test facilities are limited to fundamental studies, a more ambitious programme is under way. Present facilities include ion and electron simulation as well as the fast breeder reactor YOYO. With JT-60 under construction and EFR in the design state, fusion test capabilities will be greatly enhanced. At the present time, a more vigorous involvement is instituted between Japan and both the USA and the European Community.

J. Nihoul (SCK/CEN, Belgium) discussed progress in materials development in Europe. The European fusion materials programme is based on investigations of (1) corrosion effects, and (2) bulk radiation damage effects for structural materials applications. Plasma-materials interactions, neutronics and engineering aspects have lower priority in the European programme. While research on fundamental aspects of radiation effects is going strong in various European research centres, specific goals have been defined for fusion materials data base during 1982–1986. The development programme has given priority for the following alloy systems: (1) austenitic stainless steels; (2) martensitic/ferritic steels; (3) aluminium alloys; and (4) other alloy systems (Ni, Ti, V, and Mo). For austenitic alloys, the main tests are the in-pile fatigue creep and crack growth, helium and hydrogen effects, in-pile creep, phase stability, swelling, fracture and effects of irradiation on welds. The emphasis of the testing programme on ferritics is on the ductile-brittle behaviour, hydrogen effects, fatigue-creep, fatigue crack growth, magnetic effects, and swelling. Aluminium alloys are developed primarily in Ispra, Italy, for their potential as low-activation materials. Screening tests and development of data base for Ni, Ti, V and Mo alloys are also included.

The main projects within the European programme on fusion materials were described by J. Nihoul [1] and D. Kaletta [2] (KFA, Federal Republic of Germany). The investigations were described as being focused on candidate materials for post-JET fusion tokamaks (INTOR, NET). The main projects of associated laboratories are [2]:

1. In-pile study of fatigue and crack growth (austenitic SS)
2. In-pile study of creep (austenitic SS)
3. Out-of-pile study of fatigue and crack growth (austenitic SS and other alloys)
4. In-beam study of fatigue and crack growth (austenitic SS and other alloys)
5. In-beam study of creep (austenitic SS and other alloys)
6. Post-irradiation testing after dual-beam irradiation (all alloy systems)
7. Development of helium-doping techniques for non-nickel-containing alloys
8. Theory and modelling for intercorrelation studies
9. Irradiation testing of insulators.

The European efforts on liquid-metal corrosion studies are directed toward investigating the main problems related to the use of liquid metals (lithium and lithium-lead). Current corrosion research is performed at Mol (Belgium), Ispra (Italy), and KFA (West Germany). The main aspects of this research are [2]:

(1) Static and dynamic tests
(2) Tensile properties and fracture mode
(3) Material compatibility
(4) Corrosion parameters (temperature, time, velocity, purity, and heat flux)

(5) Effects of coatings (Mo, Ta and TiC)
(6) The role of corrosion inhibitors, such as aluminium.

Proposed testing is in the areas of multi-axial creep, fatigue and magnetic tests.

2. MAIN RESULTS

2.1. Constraints in materials selection

The interactions of plasma and fast neutral atoms with surfaces generate impurities via desorption, arcing, sputtering, blistering and evaporation. R.W. Conn (UCLA) reviewed the unique conditions imposed on special components which must be designed to handle heat, particle, and electromagnetic loadings at the plasma edge. Encouraging technological developments were reported for both new materials and new component concepts. Special coatings (TiC and TiB_2) are developed by both US (Sandia Laboratories, Albuquerque, NM) and Japanese scientists (Ibaraki and Nagoya Universities). Concepts for pumped limiters and for divertor plates are being developed, particularly as part of the INTOR and US Fusion Engineering Device (FED) activities. In-vessel components will experience high particle and heat loadings. The development of such components is a key to plasma and materials performance. The progress of materials analysis for FED was reviewed by R.W. Nygren (HEDL). A working concept has been developed for FED with superconducting coils and a burn cycle of <100 s. The nominal life of FED magnet components cycled some 350000 times in a magnetic field of 9 T was shown to be limited by fatigue crack growth. Radiation levels in the TF coils are set by a maximum exposure criterion of 10^9 rad in the organic insulator (G-10 CR). A driving design feature in FED is plasma disruptions. The reference scenario was taken as a nominal peak energy flux of 1 $MJ \cdot m^{-2}$ (times a design factor of five), to be concentrated on the inboard, top and bottom surfaces of the plasma chamber. A major uncertainty still remains regarding the energy deposition time.

M.A. Abdou (ANL) and co-workers reviewed the issues related to materials in the International Tokamak Reactor (INTOR). Teams from Euratom, Japan, the USA and the USSR contribute to this effort. The first wall is subject to a neutral wall load of 1.3 $MW \cdot m^{-2}$, a surface heat load of 11.6 $W \cdot cm^{-2}$ and a charge-exchange flux (200 eV) of $3.3 \times 10^{20}\, m^{-2} \cdot s^{-1}$. In addition to the high erosion rate by charge-exchange particles, the inboard wall has to withstand plasma disruptions. The peak heat flux during the disruption is 150 MW for 20 ms. To satisfy design criteria, three options were considered for the first wall: (1) bare aluminium wall; (2) bare stainless-steel wall; (3) radiatively-cooled graphite tiles on the inboard. Although graphite was found not to melt under disruptions, it poses problems in the areas of tile attachment and chemical sputtering at the high temperatures. A bare stainless-steel design was developed to meet all design criteria. The analysis predicts that a thin melt zone develops on the first wall during disruptions. It is uncertain, however, if the melt zone will be sufficiently stable in the presence of magnetic fields. The operating conditions of the divertor collector plate were shown to be severe. The plate, which was chosen as tungsten because of its low sputtering yield, is subject to a peak heat flux of 3 $MW \cdot m^{-2}$ and a peak ion flux of $2.2 \times 10^{22}\, m^{-2} \cdot s^{-1}$.

Emphasis was given to the importance of safety and environmental considerations in fusion materials selection. The contribution of J.G. Crocker (EG&G, Idaho) and F.W. Wiffen (ORNL) focused on a number of issues that relate to the acceptability of fusion as a power source. Both the electrical utility industry and the general public will demand that new power sources be economical, safe, reliable, and licensable. These issues are subject to some degree of control by proper materials selection and development. The primary safety and environmental concerns associated with a D-T burning fusion reactor are:

(1) the presence of radioactive material inventories
(2) energy sources and mechanisms that could potentially lead to release of radioactivity during normal operation or during an accident
(3) radioactive waste management.

The use of austenitic steels as a structural material results in the generation of intense, relatively long-lived activation products. Similarly, the use of liquid lithium for tritium breeding may be desirable from a breeding viewpoint. However, the potential for a large chemical energy release during a lithium spill could be a mechanism for release of tritium and/or activation products. The theme of the presentation was that development of improved materials with longer service life-times and minimum potential for generation and release of radioactivity should be a crucial factor in determining a materials acceptability.

2.2. Fundamental studies

Work in this area focused on:

(1) displacement cascades
(2) experiments and theory of irradiation creep mechanisms
(3) interaction between point defects and helium atoms
(4) cavity formation
(5) microstructures
(6) precipitation under irradiation.

In the following, only selected results will be presented.

In-situ creep studies under 14-MeV neutron irradiation at LLNL were reported by W.L. Barmore and co-workers. Tensile creep and stress relaxation tests were run at RTNS-II on annealed polycrystalline niobium from 500 to 600°C and stresses up to 120 MPa. Deformation rates were found to increase abruptly if the beam was suddenly flashed onto the specimen surface (3-fold). The effect of pulsed irradiation was therefore an enhancement of the magnitude of creep strain. Data analysis revealed, however, a thermally activated mechanism to be responsible for the enhancement. Other ion-irradiation creep experiments were also reported at the conference. P. Jung (KFA, FRG) outlined the need for theory-experiment correlation in this area. One straightforward test of theoretical models is the stress dependence of creep rates. 20% cold-worked alloys were irradiated by a 6.2-MeV proton beam at the Jülich compact cyclotron at 300°C. At creep rates per displacement dose above $3 \times 10^{-3}\,dpa^{-1}$, the creep rates showed a quadratic stress dependence, while at lower deformation rates, stress exponents equal to or less than unity were found. Theoretical treatments of pulsed irradiation creep by H. Gurol, N.M. Ghoniem and W. Wolfer showed the possibility of creep enhancement in pulsed fusion reactors, when the creep mechanism is the climb-controlled glide of dislocations. This may be significant from a design standpoint, since irradiation creep helps to relax swelling-induced stresses. N.M. Ghoniem and R. Schafer (UCLA) also presented recent results on pulsed irradiation effects on Inertial Confinement Fusion Reactor (ICRF) materials. A new computational technique was developed to integrate rate equations for 10^5 pulses using extrapolation methods. L.K. Mansur and W.A. Coghlan (ORNL) presented results of theoretical analysis of irradiation creep for fusion reactors. From their evaluation, two unique areas were identified as important. The increased helium production results in increased cavity nucleation and, therefore, a change in the sink character of the medium. For the case of SIPA creep, the additional cavities were found to decrease the creep-rate versus swelling-rate ratio. The second important effect in fusion was described in terms of cascade-enhanced dislocation climb. Faster creep rates may result in the fusion environment.

Helium effects on materials attracted the interest of a large group of investigators. The KFA group, for example, presented papers on helium diffusion, embrittlement and bubble formation. It was also observed by A. Van Veen (Technological University, Netherlands) and co-workers that helium 'disks' precipitate at room temperature in molybdenum.

On the theoretical side, R. Bullough (Harwell) and co-workers presented recent work on the coupling between swelling and irradiation creep in fusion materials. They also continued to develop sink strength analyses suitable for rate theory applications. K. Kitajima and co-workers (Kyushu University, Japan) focused on the effects of displacement cascades on the evolution of the microstructure, using stochastic theory.

Pulsed irradiation effects enjoyed reasonable attention from both theory and experiment. In a major review article [3], we have recently shown that pulsing effects will be more dramatic for ICF systems where the pulsing times (on/off) are of the order of the reaction times for helium, vacancies and interstitials. Other theoretical results by L. Kmetyk (Sandia) and co-workers emphasize the importance of temperature pulsing. Swelling is reduced if T_{pulse} is above the peak swelling temperature, and is enhanced if T_{pulse} is less than the peak swelling temperature. The dual-beam pulsing experiments of ORNL show complex features. Fast pulsing (0.5 s on/0.5 s off) enhances nucleation when helium is co-implanted and reduces growth as compared to slow pulsing (60 s on/60 s off). Without helium, the trend is exactly opposite. D. Kaletta (KFK), on the other hand, reported on pulsed-irradiation experiments on vanadium. Helium beam pulsing (300 s beam on/off) showed an increase of bubble swelling. Also, nickel beam pulsing showed a reduction of void swelling. Irradiation pulsing effects seem to be subtle for tokamaks, but must still be considered in design equations. Conversely, for ICFR's, the effects are pronounced and require special attention.

2.3. Structural materials

Helium effects on the swelling of irradiated steels were discussed extensively by H. Brager and F. Garner

(HEDL), and by P. Maziasz and M. Grossbeck (ORNL). Helium is found to most markedly affect the nucleation of cavities. The levels of swelling and swelling rate dependence on helium content are still not very clear. The most important effects of helium are an enhancement of both the low- and high-temperature swelling rates as compared to the case with little helium. The intermediate temperature range is still a matter of interpretation, since the data base is not adequate.

Strength/ductility correlations were presented by R.L. Simons (HEDL) for 20% cw AISI 316 SS. A high-exposure softening was observed in the temperature range $300 < T < 500°C$. This was attributed to cavity formation and the indirect effect on reducing the dislocation network. HFIR specimens showed more softening than the EBR-II specimens owing to the higher helium content. A correlation equation was characterized by an initial irradiation hardening which saturates, followed by a softening effect. This effect was correlated to the square root of the product of helium concentration and displacement per atom.

Titanium-modified austenitic stainless steels were reported by the ORNL group to be resistant to high-temperature irradiation and helium effects. A fine dispersion of titanium carbide precipitates controls the cavity size distribution, leading to a reduction in the amount of observed swelling. The role of second-phase particles on the growth of cavities was adequately shown by the experiments of E.H. Lee and co-workers (ORNL). Interfaces between the G, η, and Laves phases and the matrix of irradiated steels provide sites for the development of large cavities. The growth of cavities was found to be associated with the growth of precipitates. P. McConnell (Fracture Control Corp.) and co-workers outlined procedures for assessment of fracture problems in fusion structures. Fracture of irradiated materials can be studied in three different modes of fracture failure:

(1) failure by unstable crack extension
(2) general yielding failure by plastic collapse
(3) transition regime where elastic plastic behaviour occurs.

Methods using Linear Elastic Fracture Mechanisms (LEFM), and J-integral were suggested. The subject of flow and fracture in the fusion environment was addressed by W. Wolfer (Univ. of Wisconsin) and R. Jones (PNL), with particular emphasis on the slow growth of fatigue cracks in Stages I and II of the crack growth relationship. Applying a model developed by Smith et al., for flow localization, they predict that highly irradiated 304 SS will have a satisfactory fracture toughness ($\geqslant 27$ MPa·m), even though its ductility is practically nil in a tensile test.

Innovative concepts for materials development are pioneered by the MIT and ORNL groups. The potential for using rapid solidification for improved irradiation performance is developed by O.K. Harling et al. (MIT). Rapid solidification is achieved by cooling rates greater than 1000 $K \cdot s^{-1}$. Solidification must be followed by compaction to produce bulk material, since at least one dimension will be small after solidification. The very fine and stable dispersion of precipitate particles ($10^{21} - 10^{23} \cdot m^{-3}$) can nucleate a very fine dispersion of helium bubbles. This may reduce swelling. The development of Fe-base long-range-ordered (LRO) alloys for fusion reactor applications was reported by C.T. Liu (ORNL). The solute atoms in LRO alloys arrange themselves periodically and form an ordered crystal structure. This offers potential advantages over conventional alloys. Kinetic processes involving solid-state diffusion such as creep and migration of point defects are reduced in the ordered lattice. Also, the unique dislocation dynamics of ordered alloys provides excellent high-temperature strength and fatigue resistance. The reported work on the $(Fe, Co, Ni)_3V$ system showed that the brittle limitation in the ordered state can be overcome, with tensile elongation exceeding 35% at room temperature.

F. Garner and co-workers (HEDL) presented a number of papers related to the structural aspects of steels under irradiation. Yield strength measurements were correlated to microstructural features such as dislocations, loops, voids, and precipitates. It was reported that both annealed and cold-worked materials approach the same saturation level of the yield stress at a fluence of $(2-3) \times 10^{22} n \cdot cm^{-2}$ ($E > 0.1$ MeV). While the saturation behaviour was explained to be independent of almost all irradiation and material variables (microstructure, displacement rate, temperature, stress, and helium level), it is interesting to compare with the results of R.L. Simons (HEDL), and with F.W. Wiffen and P. Maziasz (ORNL). Simons reported a softening of the yield strength after the saturation behaviour of high fluences, which he correlated to the square root of the product of helium concentration and displacements per atom. ORNL irradiations of cw 316 SS at 55°C showed modest increases in the yield strength at 10 dpa (15%). Maziasz and Grossbeck presented irradiation results of types 316 and (316+Ti) austenitic stainless steels irradiated in HFIR. Their work was focused on the swelling and microstructural features up to 60 dpa, and in the temperature range of 55 to 670°C. It was pointed out that the amount of swelling in cw (316+Ti) is less than for cw 316. This was attributed to the formation of TiC precipitates with their influence on the clustering of helium-vacancy complexes, and on dislocation pinning.

Ferritic alloys are becoming more of a focus for fusion reactor studies. One major reason is the low swelling rate of this class of alloys. An important finding reported by D. Gelles (HEDL) was a minimum in the swelling rate as a function of the chromium content around 3% Cr. This confirms earlier findings by Little and co-workers (Harwell). It was concluded that void swelling in ferritic alloys will not represent a design limitation for fusion applications. Extensive precipitation was observed by Gelles, and it was conjectured that mechanical properties will therefore be affected.

2.4. Plasma/materials interactions

The behaviour of plasma ions and neutral atoms in the close vicinity of tokamak reactor inner walls and in-vessel components is of critical importance. First, such components which are subject to a high energy and particle flux may undergo sputtering erosion, blistering, evaporation, and general degradation of mechanical and physical properties. Second, impurity ions introduced into the plasma generally contribute to radiation energy losses. Papers addressing these questions ranged from theoretical calculations of relevant phenomena such as sputtering and erosion to the more practical aspects of fabrication technology of coated surfaces. Hydrogen re-cycling properties of stainless steels were reviewed by K. Wilson (Sandia, Livermore). Three main mechanisms for hydrogen release from stainless-steel first walls were discussed. These are reflection, ion desorption and molecular recombination. It was noted that the reflection coefficient of hydrogen ions of low energies (below 100 eV) is still to be determined. While experiments are difficult to perform, theoretical calculations are hampered by the breakdown of the binary-collision approximation and by the need to incorporate attractive potentials. Another area of major uncertainty is tritium retention in structural materials subject to tritium surface bombardment. Unfortunately, present calculations and measurements for the hydrogen recombination rate on stainless steels near room temperature vary by five orders of magnitude. This range of values can have an enormous effect on the predicted re-cycling and tritium inventory.

Sputtering yield calculations for deuterium incident at inclined angles were reported by W. Eckstein (Max-Planck-Institut, FRG), and by J.P. Biersak (Hahn-Meitner Institut, FRG) and L.G. Haggmark (Sandia, Livermore). Their calculations were based on a modified version of the binary-collision, Monte-Carlo code TRIM. For neutral-beam applications, the calculations were performed up to ion energies of a few hundred keV and angles of incidence of up to 88° to the surface normal. It was found that the sputtering rate increases by about an order of magnitude for such inclined beams. The maximum of the sputtering yield of deuterium shifts to a higher energy of 80 keV. Deuterium diffusion measurements were reported by Tanabe and co-workers (Osaka University, Japan). Their results indicate that deuterium diffusion is not classical (ln D versus 1/T linear), but is controlled by the damage zone created at the front surface. Diffusion coefficients deduced from re-emission studies were a factor of 10–100 smaller than measured deuterium diffusion coefficients due to trapping of deuterium on defects. High hydrogen desorption cross-sections from steel by ion impact ($(5-10) \times 10^{17}\,cm^{-2}$) were reported by R. Bastasz (Sandia, Livermore). The resulting cross-section data made it possible, given the energy spectrum of the hydrogenic flux striking the first wall, to calculate the efficiency of particle impact in removing adsorbed hydrogen.

2.5. Compatibility between liquid metals and structures

Chemically compatible blankets and coolants are desirable to reduce the degradation of the physical and mechanical properties of blanket structures. Many designs for fusion power reactors utilize liquid lithium as the energy conversion and tritium breeding medium. Several methods for reducing corrosion by lithium were reported by G.C. Burrow and co-workers (Westinghouse R and D Center, USA). High-temperature gettering of non-metallic impurities or the use of soluble additives (aluminium or calcium) were reported as examples. An aluminizing process was used to coat the surface of various austenitic and ferritic steels with aluminium. Good compatibility was obtained at 400°C and $\Delta T = 10°C$, without loss of material or degradation of properties. Nitrogen impurities were noted to affect the corrosion rate. For an impurity level of about 100 ppm, it was found that lithium corrosion produces rapid surface chemistry changes, independent of velocity below $2\ m \cdot s^{-1}$ and downstream position.

The Ispra (Italy) group (F. Brossa and co-workers) reported on molybdenum coating of stainless steel and Inconel alloys for improved resistance to lithium corrosion. Molybdenum was sprayed with a plasma arc gun under argon and hydrogen atmosphere. The Mo surface coating has the potential of eliminating the nickel leaching problem from stainless steels, especially where nitrogen impurities are present. V. Coen and co-workers (Ispra) presented results of experiments on the corrosion of 316 SS with $Li_{17}Pb_{83}$ eutectic. This material, which melts at 508 K, has the potential advantages of high breeding ratio, minimum tritium inventory and minimum hazard potential in contact with water or air. Again, the major problems were identified as selective leaching of nickel, possible loss of ductility and fracture strength, and stress corrosion cracking.

Fatigue tests of HT-9 and type 304 SS in a flowing lithium environment were conducted by O. Chopra and D. Smith (ANL). At levels of <500 wppm, nitrogen at 755 K, fatigue life was reported to be similar to the case of a steel-sodium system. However, for nitrogen impurity levels of >1000 wppm, fatigue life was lower by a factor of 2–5.

2.6. Solid breeder materials

The primary function of the solid breeder in a fusion reactor is to generate tritium which can subsequently be used for fuel. The candidate material must possess both a high lithium atom density and a low tritium inventory. D. Suiter and J. Davis (McDonnell Douglas Astronautics Co., St. Louis, USA) reviewed the potentials of candidate compounds (Li_2O, $LiAlO_2$, Li_4SiO_4, Li_2SiO_3 and Li_2ZrO_3). It was shown that for a reasonably rapid release of tritium, a fine-grained ($\leqslant 2$ μm size) porous compact (60–70% of theoretical density) is needed. The possibility of radiation sintering is being determined at HEDL. A technique that was explored by McDonnell Douglas Co. to overcome radiation-enhanced sintering is to introduce pressed organic fibres into the compact. The compact is then heated until the fibres are burned out, leaving clear channels as pathways for tritium release.

Interfacial compatibility studies of selected ceramic tritium breeder materials and typical structural alloys were presented by P. Fin and co-workers (ANL), and by O. Chopra and D. Smith (ANL). Their results show that Li_2O is the most and $LiAlO_2$ the least reactive of the breeding materials in contact with structural components. The development status of solid breeder materials was reviewed by C. Johnson (ANL) and G. Hollenberg (HEDL). It was pointed out that Li_2O is extremely attractive because of the high lithium atom density. However, the solubility of T_2O in Li_2O, forming LiOT, may act against ease of tritium recovery. During breeding, T_2O will form and, in combination with the solid breeder, could present a very corrosive environment. It was clear from the papers in this area that much data need yet to be collected.

2.7. Special-purpose materials

Special-purpose materials are required for many parts of the fusion system. These include organic and ceramic electrical insulators, thermal insulators, and magnet materials. Electrically insulating materials are required for such applications as:

(1) windows in RF heating systems
(2) neutral-beam injector insulators
(3) the toroidal-current break
(4) magnetic-coil insulators
(5) insulators for direct convertors.

Thermal insulators will function in a high-neutron-irradiation environment for the insulation high- and low-temperature components.

F.W. Clinard (Los Alamos National Laboratory, USA) summarized the work on ceramic and organic electrical insulators for fusion applications. Ceramic windows for radio frequency (RF) energy transmission will be subjected to high power loading ($5\times10^4\,kW\cdot m^{-2}$). If energy absorption is too great, thermal stresses will fracture the window. BeO and Al_2O_3 are candidate materials for window usage, although little is known about the effect of irradiation on the loss tangent of these ceramics at RF heating frequencies. Neutral-beam injector systems require electrical insulation in the ion source to resist large electric fields ($1\,kE\cdot mm^{-1}$): Al_2O_3 and MACOR glass-ceramic (product of Corning Glass Works). Toroidal current breaks are used to interrupt the electrical continuity of the metallic structure. Electrical requirements are modest, and therefore ceramics such as Al_2O_3, $MgAl_2O_4$, Si_3N_4 and SiC were suggested as candidate materials. Superconducting toroidal field magnets can make use of organic insulators if shielding is adequate. Epoxies and polyimides are prime candidate materials. Direct convertors will require insulators capable of holding off a significant fraction of the several-keV-particle voltage. Ion and X-ray fluxes could be significant. Electrolysis (separation of cations and anions in a DC field) could result in severe degradation if insulators work at high temperatures. High-quality refractory forms of ceramics such as Al_2O_3, MgO and $MgAl_2O_4$ must be used.

R.A. van Konynenberg and co-workers (Lawrence Livermore Laboratory, USA) conducted a series of experiments at RTN-II to irradiate superconductors and magnet stabilizers. Samples of NbTi, copper and aluminium were irradiated with 14.8-MeV neutrons at 4 K. The critical current of NbTi and the electrical resistance of Cu and Al were measured in transverse fields of up to 12.4 T. Annealing at room temperature and re-irradiation were then performed. After a fluence of $8 \times 10^{20} n \cdot m^{-2}$, the maximum observed change in the NbTi critical current was a decrease of 3% at 4 T. At higher magnetic fields there were no significant changes from the irradiated values. C.L. Snead (Brookhaven National Laboratory, USA) and co-workers reported on experimental studies of high-energy neutron damage in Nb_3Sn, using RTNS-II.

2.8. Test methods

Because of the limited test volume available for high-energy (14 MeV) neutron irradiations in the near term, a need exists for test techniques which can be used to extract mechanical property data from small samples. G. Lucas and co-workers (Univ. of California, Santa Barbara, USA) reported on new techniques to measure the ductility of thin sheet samples. Sheet metal forming techniques such as the shear punch test and the bulge test were reported. The failure displacement was correlated to the total ductility. R. Blewer and co-workers (Sandia, Albuquerque, USA) discussed recent experimental results for materials testing and development for designing commercial-path reactor components in machines such as FED. High test and particle flux experiments were conducted on composites of low-Z coatings on structural alloys. Thermal loads were varied from 22 $kW \cdot cm^{-2}$ for a few milliseconds to 4 $kW \cdot cm^{-2}$ for 100 hours. Critical heat flux calculations indicated that burnout will occur at a power density of 10 $kW \cdot cm^{-2}$. Thermohydraulics, thermal fatigue and internal/external erosion were addressed in these experiments. Non-destructive monitoring at various points during the thermal fatigue testing were performed and post-mortem analysis after 10000 heat cycles was accomplished using the Rutherford ion backscattering technique.

E. Opperman (HEDL) described the methods of experimentation in FMIT. It was pointed out that during irradiation, experimental hardware becomes highly activated and subsequent handling has to be done totally remotely. Miniaturized fatigue crack growth specimen technology and results were reported by R.J. Puigh and co-workers (HEDL). The miniature fatigue crack growth technology uses a small, centre-notched specimen which is fabricated from their sheet stock material. This smaller specimen size was reported to permit 120 of the miniature specimens to occupy the same irradiation volume that one standard-size specimen would occupy. An electrical potential technique was developed to measure the crack extension during cyclic loading. F. Huang and G. Wire presented recent results of fracture toughness testing on ferritic alloys using the electropotential technique.

3. SUMMARY

Materials development and research are recognized as a key to the development of a practical fusion reactor. Significant progress has been achieved in the United States, Japan, and the European Community. For the near term, greater emphasis is necessary in areas such as plasma-materials interactions and surface physics. Moreover, a wider data base on neutron-irradiated bulk materials is central to the development of attractive reactor concepts. Sustaining and enhancing the materials programme must remain a central tenet of fusion development strategy.

ACKNOWLEDGEMENTS

This work is supported by the US Department of Energy, Office of Fusion Energy, under Contract No.4-482530-25106.

REFERENCES

[1] Proc. 2nd ANS Top. Meeting on Fusion Reactor Materials, J. Nucl. Mater. (in press).
[2] KALETTA, D., Proc. of SMIRT-6 Post-Conference Seminars on Fusion Structural, Paris, France (August 1981).
[3] GHONIEM, N.M., KULCINSKI, G.L., Nucl. Technol. 2 2 (1982) 165.