Proceedings of the Fourth International Symposium
on
ADVANCED NUCLEAR ENERGY RESEARCH
ROLES AND DIRECTION OF MATERIAL SCIENCE IN NUCLEAR TECHNOLOGY

ADVANCED MATERIALS — THE KEY TO ATTRACTIVE MAGNETIC FUSION POWER REACTORS*

Everett E. Bloom
Oak Ridge National Laboratory
P.O. Box 2008, Oak Ridge, TN 37831-6117
Phone 615-574-5053, FAX 615-574-7659

ABSTRACT

Fusion is one of the most attractive central station power sources from the viewpoint of potential safety and environmental impact characteristics. Studies also indicate that fusion can be economically competitive with other options such as fission reactors and fossil-fired power stations. However, to achieve this triad of characteristics we must develop advanced materials with properties tailored for performance in the various fusion reactor systems. This paper discusses the desired characteristics of materials and the status of materials technology in four critical areas: (1) structural materials for the first wall and blanket (FWB), (2) plasma-facing materials, (3) materials for superconducting magnets, and (4) ceramics for electrical and structural applications.

KEYWORDS
Fusion Reactors, Alloy Development, Radiation Damage, Austenitic Steels, Martensitic Steels, Vanadium Alloys, SiC/SiC Composites, Ceramics, Magnet Materials

1. INTRODUCTION

The realization of fusion as an energy source is a major scientific and technological challenge. The potential return on the investment required to develop this energy source is, however, very large. As observed by Holdren[1], the era of cheap energy (but energy that has a significant detrimental effect on our environment) is coming to an end. We must undertake a transition to energy sources that will have reduced environmental impact. At the same time the demand for energy will continue to increase. Holdren's "optimistic projections" suggest that total energy use in the year 2060 will double that of 1990. The long-term options include solar, wind, ocean heat, biomass, geothermal, advanced fission reactors, and fusion reactors.

In the fusion reaction, nuclei of light elements, such as isotopes of hydrogen, combine to form heavier elements and, in the process, release large amounts of energy in the form of energetic particles whose kinetic energy can be converted to heat. The fusion reaction that is easiest to attain is that of deuterium and tritium. This reaction produces 17.6 MeV, in the form of a 3.5 MeV alpha particle and a 14.1 MeV neutron. For this reaction to occur at sufficiently high rates to produce useful energy, a mixture of deuterium and tritium gas must be heated to temperatures in excess of 150×10^6 °C. At this temperature the deuterium and tritium atoms are ionized, forming a plasma consisting of the atomic nuclei and electrons. The plasma is confined by magnetic fields within a vacuum chamber. Deuterium is readily available in nature. Tritium, however, is an unstable isotope of hydrogen, having a half-life of about 12 years, and thus does not occur naturally. In a fusion reactor, tritium will be produced in lithium by capturing the neutrons produced in the fusion reaction. The following reactions apply: $Li^6 + n \rightarrow He^4 + T$ and $Li^7 + n \rightarrow He^4 + T + n$.

Figure 1 is a schematic drawing showing the basic components of a magnetically confined fusion reactor core. Progressing outward from the plasma is the first wall of the vacuum chamber and divertor. The divertor serves to remove helium from the plasma. The 3.5 MeV carried in the alpha particle or about 20% of the energy of the fusion reaction must be removed as a heat flux through the surface of the first wall and divertor. The blanket structure has two primary functions: recovery of the 14.1-MeV energy contained in the neutron and containment of the breeding media that contains lithium (e.g., liquid lithium, a lithium ceramic, or a lithium salt) for producing tritium. The blanket and shield together serve to reduce neutron and gamma radiation levels at the torrodial field (TF) magnets and beyond. The TF magnets produce the magnetic fields which contain the plasma. For reasons of efficiency, the TF coils will be superconducting. Beyond the fusion core are systems for converting the heat of the fusion reaction into electricity or for using it as process heat, and for recovering and recycling tritium.

Like nuclear fission, fusion is an energy source with high specific energy. It is only suited for central station power

*Research sponsored by the Office of Fusion Energy, U.S. Department of Energy, under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

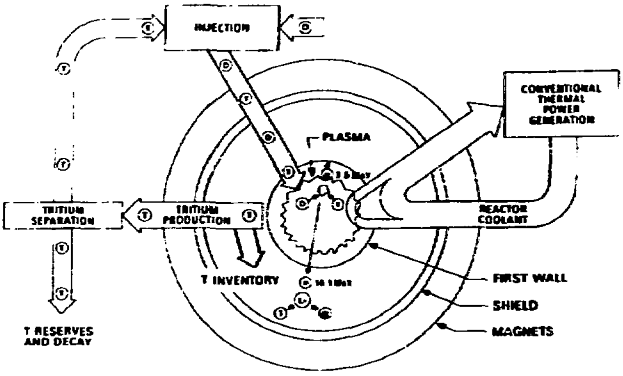

Fig. 1. Schematic of Fusion Reactor with Tritium Recovery and Thermal Conversion Systems.

generation. Relative to fission, fusion offers potential advantages in the critically important areas of safety and environmental impact. The fusion reaction does not produce radioactive nuclides, and the reaction can only be achieved and maintained by containing pure plasmas with strong magnetic fields. Should the magnetic fields decrease, the plasma is quenched. If the plasma is contaminated, such as would occur by a leak of coolant into the vacuum vessel, the plasma will be quenched. Any radioactivity produced in the fusion reactor derives from the capture of neutrons in the materials from which the reactor is constructed. The characteristics of this induced radioactivity (i.e., type, half-life, biological hazard potential, decay heat, pathways for dispersal, etc.) are dependent on the chemical composition of these materials. The full potential safety and environmental advantages of fusion can, however, only be realized by developing materials that are optimized from the viewpoint of safety (e.g., mechanical and physical properties that yield low probability of failure, reduced afterheat, reduced volatility) and environmental impact (e.g., reduced long-half-life radionuclides for safer disposal and/or possible recycle). For fusion to be economically competitive these advanced materials must have additional attributes, such as adequate service life, temperature capability to provide a thermodynamically efficient design, attractive cost, etc.

Development of materials with properties tailored to meet the challenges presented by the fusion environment involves four major steps:

- Development of a mechanistic understanding of the behavior of materials in the fusion environment;
- Demonstrating feasibility of candidate material systems based on the inherent physical, chemical, and mechanical properties, response to the fusion neutron, chemical, and mechanical environments, and positive assessment of the potential to develop a suitable material;
- Development of the candidate material systems through composition and microstructure control to achieve the required mechanical properties, radiation damage resistance, etc.;
- Development of a materials database sufficient to support engineering design, safety, and performance analysis.

This paper presents an overview of the present status of materials technology in four critical areas: (1) structural materials for the first wall and blanket (FWB), (2) plasma-facing materials, (3) materials for superconducting magnets, and (4) ceramics for electrical applications.

2. FIRST WALL AND BLANKET (FWB) STRUCTURAL MATERIALS

Figure 2 is a drawing of the International Thermonuclear Experimental Reactor (ITER) showing the location of key components and systems[2]. ITER is an experimental reactor that will generate about 1,000 megawatts from the DT fusion reactions. ITER will be operated in two stages. In the first stage, lasting 6 to 8 y, plasma physists will attempt to achieve an ignited plasma similar to that required in a fusion power reactor. During the second stage the reactor will serve as a technology development test bed. ITER will not produce electricity. From the ITER design one can begin to appreciate that fusion power reactors will be large and complex.

The FWB structure surrounds the plasma and defines the shape of the plasma chamber. The FWB structure is actively cooled and subjected to intense neutron radiation and heat fluxes. The structural materials must possess a high level of dimensional stability and mechanical integrity. Failures will be difficult to repair once the reactor is placed in service and becomes radioactive. Additional requirements include: (a) low hydrogen diffusivity and solubility to minimize tritium mobility and inventory; (b) favorable activation characteristics with regard to decay heat, safety, and waste management, and (c) ease of production, forming, and welding.

The intense radiation environment in the region of the FWB structure creates a unique challenge. Figure 3 illustrates the creation of damage through the interaction of

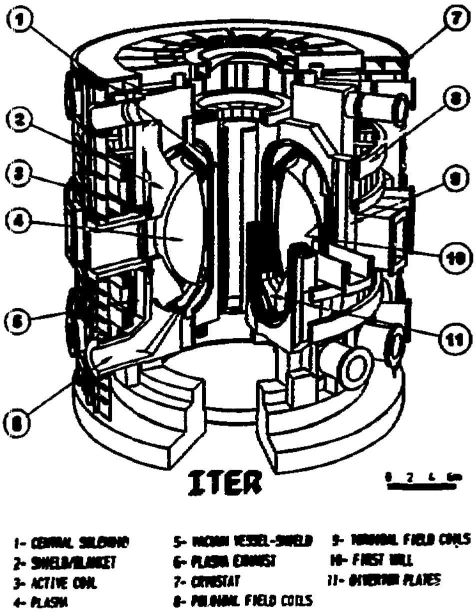

Fig. 2. The International Thermonuclear Experimental Reactor (ITER) Showing Key Components of the Fusion Reactor.

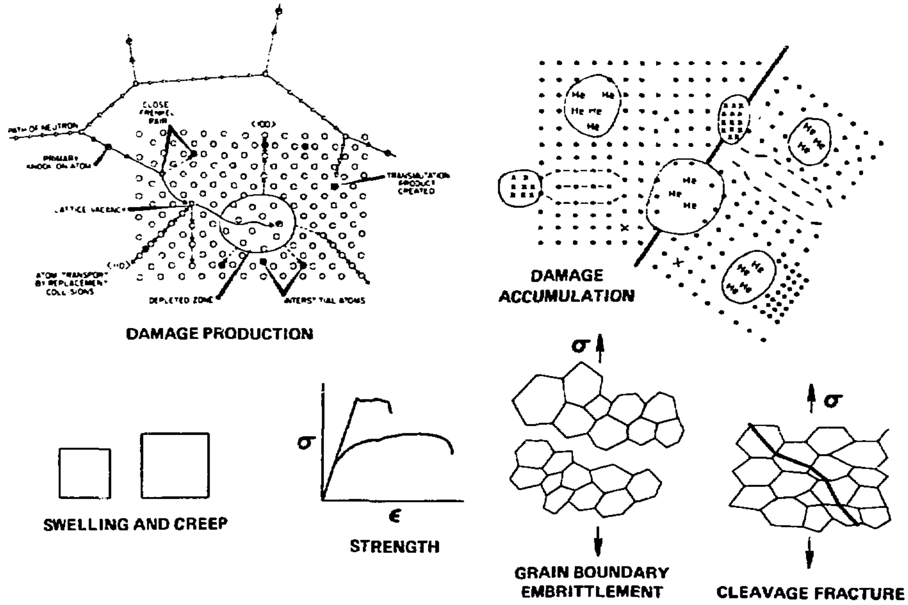

Fig. 3. Schematic Representation of Evolution of the Microstructure and Microchemistry of Complex Alloys During Irradiation the Resultant Effects on Some Important Engineering Properties of Structural Materials.

neutrons with the lattice atoms of a structural material, the evolution of the irradiation-produced microstructure, and some of the physical and mechanical properties that can be affected by the irradiation-produced microstructure. This illustration is for structural alloys irradiated between about 0.3 and 0.6 of the absolute melting temperature — the temperatures at which most alloy systems would be used in fusion applications. The structure is subjected to a broad spectrum of neutron energies ranging from 10^{-7} to 14.1-MeV. The 14.1-MeV neutrons from the fusion reaction lose their energy in the FWB structure through a series of atomic collisions. The nature of the interaction between the neutrons and the atoms of the structural material is highly energy dependent and results in three types of changes: (a) the formation of point defects (interstitials and vacancies) by the displacement of atoms from their lattice sites, (b) generation of gaseous and solid transmutants, and (c) induced radioactivity.

It is the simultaneous generation of displaced atoms and transmutant helium and hydrogen that alters the microstructure, mechanical behavior, and dimensional stability of structural alloys. Collisions with high-energy neutrons produce primary knock-on atoms (PKAs) with energies to about 100 keV. The recoiling PKAs create cascades of additional displacements. Almost 90% of these displaced atoms return to lattice sites within $\sim 10^{-13}$s. The remaining defects survive as isolated or clustered vacancies and self-interstitials. Mobile vacancy and interstitial defects migrate to pre-existing sinks, such as grain boundaries and dislocations, or cluster to form a variety of extended defects such as cavities or dislocation loops. Point defects may also annihilate by recombining with the opposite type. During the migration process, point defects diffuse to sinks and alloy constituents diffuse away from sinks. Since, in general, the constituents of an alloy will have different diffusivities, local regions near sinks can become enriched or depleted in solutes. This phenomenon of radiation-induced segregation (RIS) can drive a system far from thermal equilibrium and lead to the development of unexpected phases.

With the continuous formation of cavities and loops and the dissolution and formation of phases, the microstructure evolves to a quasi-steady-state, characteristic of the radiation environment. Extended defects are effective barriers to dislocation motion and give rise to radiation hardening.

The formation of helium through nuclear transmutation plays an important role in microstructural development and property changes. Helium atoms stabilize small vacancy clusters, which then form cavity nuclei rather than collapsing into loops. These cavities correspond to the atoms that have been knocked out of lattice sites, forming interstitials which are then subsequently absorbed to form new planes of atoms (interstitial loops). By absorbing sufficient helium atoms and vacancies, cavities frequently exceed a critical size beyond which they undergo continuous growth by vacancy absorption. This phenomenon of void swelling can result in volumetric changes of the order of tens of percent at high neutron exposures.

Helium atoms are also trapped at grain boundaries, forming small bubbles. This phenomenon, coupled with solute segregation and radiation hardening, may lead to grain-boundary embrittlement.

If stress is applied during irradiation, dislocations having different relative orientations to the stress axis absorb slightly different ratios of interstitials to vacancies. This asymmetry leads to dislocation climb, which produces creep either directly or by enabling dislocations to slip from an obstacle. This phenomenon of irradiation creep can exceed normal thermally activated creep by orders of magnitude at temperatures below about half the absolute melting point.

FWB STRUCTURAL MATERIALS FOR ITER

The presently favored FWB concept for the ITER is a water-cooled austenitic stainless steel structure operating in the range of 100 to 350°C with a maximum surface heat flux of $\sim$0.3 MW/m^2 and a coolant pressure of $\sim$2 MPa. Maximum cyclic strain due to thermal stresses are expected to be about 0.2%. The system is required to sustain $\sim 10^5$ burn cycles and a total of $\sim$30 dpa (displacements per atom).[2] The alloy composition will probably be close to AISI 316 austenitic stainless steel. There is a well established fabrication and joining technology for this alloy as well as an extensive irradiation performance and corrosion data base for temperatures >250°C. However, the ITER environment embraces a significantly different regime of neutron spectrum, flux, and temperature. Recent fission reactor experiments designed to simulate this environment have defined the primary radiation effects, which must be accounted for in assessing performance limits.[3]

Radiation hardening from loops and cavities is probably the dominant phenomenon with the increase in yield stress saturating with neutron dose after $\sim$10 dpa. The magnitude of the increase is strongly dependent on the initial microstructure and irradiation temperature. In annealed 316 SS the increase ranges from $\sim$400 MPa at 60°C to a maximum of $\sim$600 MPa at $\sim$350°C before declining sharply at higher temperatures. This phenomenon is accompanied by a loss in work-hardening capacity; uniform elongation is reduced to <1% over the range of 150 to 350°C. However, the material retains substantial capacity for plastic deformation, and tensile fracture modes are ductile-transgranular. Recent experiments have also shown that irradiation creep will play an important role in the performance of the FWB. This stress-dependent phenomenon is independent of temperature over the range of 200 to 600°C and provides an important mechanism for stress relaxation. At lower temperatures, a different mechanism prevails based upon the transient interstitial population, and creep rates are sometimes higher by a factor of 3 to 4. At sufficiently high stresses this phenomenon could lead to significant levels of deformation. For example, at 60°C a stress of $\sim$200 MPa could produce 1% creep strain after $\sim$10 dpa. It is clear that void swelling will not be a concern at temperatures below 350°C. Because of the high helium production rate (He/dpa ratio $\sim$15 appm/dpa), volumetric swelling could reach 2 to 3% in regions operating at 400°C in the lifetime of ITER. Postirradiation strain-controlled fatigue data indicate that although irradiation hardening can lead to some reduction in fatigue life, there is an endurance limit for irradiated AISI 316 of ~0.15%.

In water-cooled fission reactors, irradiation-assisted stress corrosion cracking (IASCC) is a well documented mode of environmental degradation. Radiation-induced segregation at temperatures >250°C results in grain-boundary compositional changes involving accumulation of Si, P, and S and depletion of Cr. Under suitable conditions of water chemistry and stress, grain-boundary cracking may occur. Recent experiments, however, have failed to detect any changes in grain-boundary chemistry for irradiation temperatures of 200°C and below. Consequently, IASCC is unlikely to be a major limiting factor in FWB performance.[4,5] Control of water chemistry is essential to prevent corrosion phenomena.

Although the performance of AISI 316 under these conditions is promising, several radiation-related issues remain to be clarified before defining final composition limits and obtaining a comprehensive engineering design data base. The radiation-induced changes in microstructure, which cause hardening, will also lead to changes in fracture toughness that need to be defined through irradiation experiments utilizing compact tension specimens. Another unexplored phenomenon is that of sub-critical crack growth induced by transmutation-produced hydrogen. Susceptibility to this phenomenon is related to austenite stability and represents another reason why the final material composition, particularly with regard to carbon and nitrogen levels, needs to be carefully evaluated.

<u>FWB Structural Materials for Fusion Power Reactors</u>

The Senior Committee on Environmental, Safety, and Economic Aspects of Magnetic Fusion Energy (ESECOM) study[6] indicates a serious degradation in economic performance if the fusion reactor blanket lifetime falls below 10 MWy/m^2. The cost of electricity decreases with increasing blanket life between 10 and 20 MWy/m^2 and becomes relatively insensitive to life beyond 20 MWy/m^2. Developing FWB structural materials having a lifetime capability of 20 MWy/m^2 represents a significant challenge, possibly a greater challenge than development of materials for any other energy system. The operating environment is severe, and the resultant number of mechanical and physical properties that must be simultaneously optimized is very large, as illustrated in Table 1. Safety considerations and minimization of environmental impact severely limit the number of alloy or ceramic systems in which suitable structural materials might be developed. From the viewpoint of environmental impact, it is clearly desirable to minimize or eliminate elements that will develop long half-life radioactivity in service. The United States Fusion Materials Development Program has used as a guideline or target the development of materials that could be disposed of by shallow land burial within the intent of "Nuclear Regulatory Commission, Licensing Requirements for Land Disposal of Radioactive Waste, 10CFR, Part 61, of the Federal Register." It is recognized that these requirements

Table 1. Many Properties are Important in the Performance of Materials in Fusion Reactors Because of the Complex Operating Environment

Operating Environment	Material Response/Issues
Radiation Damage Displacement Transmutations	Swelling Irradiation Creep Degradation of Physical and Mechanical Properties
Chemical Compatibility Coolants Tritium Breeders	Corrosion - Mass Transfer Degradation of Mechanical Properties Hydrogen Embrittlement
Elevated Temperatures	Time Dependent Deformation (Creep)
Mechanical Stresses Primary Secondary (Thermal) Cyclic High Loading Rates (Disruptions)	Many Mechanical Properties Critical in Design Tensile Fatigue Crack Growth Fracture Toughness
Complex Structure	Fabricability Welding/Joining Maintenance

were not written for the disposal of radioactive materials from fusion reactors and do not even cover all elements that would be found in such a waste stream. To meet these guidelines radioactivity must decay to acceptable levels within 500 y (acceptable level is defined to mean that inadvertent entry into the waste, with continuous occupancy, would result in less than 500 mrem yearly dose to the intruder). Figure 4 shows the decay of radioactivity in pure elements irradiated in a fusion spectrum for 9 MWy/m^2. A goal of minimizing long half-life radioactivity will eliminate or severely restrict the use of many elements, such as Nb, Mo, Al, N, Cu, and Ni.

The factors involved in optimization of fusion for safety are complex. The probability, propagation, and consequences of a failure are primary considerations. Fusion

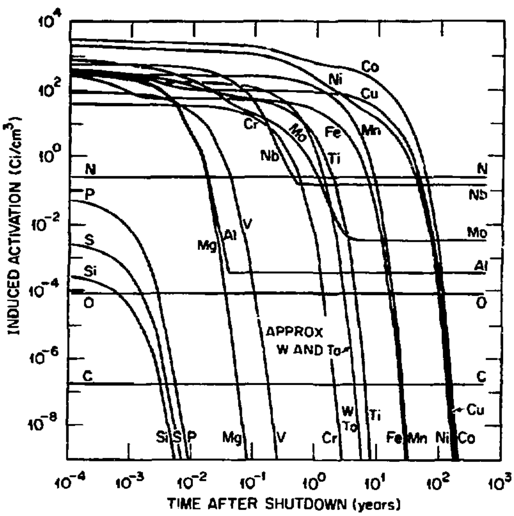

Fig. 4. Calculated Activation and Decay of Pure Elements Irradiated in a Fusion Reactor Neutron Spectrum for 9 MWy/m^2.

reactor designs have not been carried out in sufficient depth to assess accident probabilities and propagation, thus, material properties that influence these characteristics have not been clearly identified and incorporated into material development activities. At present, minimization of the consequences of an accident have been the primary input from the safety/design community to the materials effort. From the viewpoint of accident consequences critical material properties include afterheat from radioactive decay, pathways for dispersal of radioactivity, and radiological hazard.

For power reactor applications, international programs are currently evaluating ferritic-martensitic (FM) steels containing 9 to 12% Cr, vanadium-based (V) alloys, and silicon carbide-silicon carbide composites (SiC/SiC). Three structural material options are pursued for two basic reasons. First, none of these three options has the physical, mechanical, and chemical properties to make it a viable candidate in all design options (i.e., coolant-breeder combination, operating temperature, wall loading, etc.). Ferritic-martensitic steels are most attractive in a water-cooled, ceramic-breeder system. Vanadium alloys are the most attractive candidate for a Li or Li-Pb coolant/breeder and SiC-SiC composites are proposed for very high temperature helium cooled designs. Secondly, three options are being pursued because it is not clear which of the options can be developed to have adequate properties for fusion power reactor applications. Table 2 lists the attractive properties/characteristics, major feasibility issues, and design issues for the three advanced materials and compares them to the austenitic stainless steels.

Table 2. Attractive Characteristics, Feasibility Issues, and Limitations of Proposed Alloy and Ceramic Systems for Structural Applications

Structural Materials	Attractive Characteristics	Major Feasibility Issues	Design Issues/Limitations
Ferritic Steels	Swelling and helium embrittlement resistance	Irradiation effects on fracture properties	Upper temperature limit
Austenitic Steels	Established technology Compatibility and mechanical properties	Irradiation induced swelling	Activation Thermal stress
Vanadium Alloys	Temperature capability Swelling and helium embrittlement resistance Thermal stress factor	Irradiation effects on fracture properties	Chemical compatibility
SiC/SiC Composites	Low activation Temperature capability	Radiation effects on physical and mechanical properties	Design methods Cost

Ferritic steels exhibit excellent swelling and helium embrittlement resistance at irradiation damage levels typical of a fast reactor. Design studies have indicated that ferromagnetic alloys would be acceptable as the structural material in a magnetically confined fusion reactor. The two most critical remaining questions on the development and use of this alloy system relate to (1) the effect of increased helium generation rate in the fusion spectrum on radiation response and (2) the effect of irradiation on the fracture properties. Substantial progress has been made in addressing these questions.

Helium effects have been studied with ion irradiations and by doping commercial FM steels such as 12Cr-1MoVW (HT-9) and 9Cr-1MoVNb with small quantities of nickel and irradiating them in fast reactors, in which very little helium is produced, and comparing the results with the same steels irradiated in mixed spectrum reactors, where the fusion He/dpa ratio is produced. Displacement damage and helium appear to have less effect on most of the properties of ferritic steels than on the austenitic steels. Fast reactor irradiations of ferritic steels have been conducted to damage levels in excess of 100 dpa. Although void nucleation, growth, and the resultant swelling do occur during fast reactor irradiation, the incubation times are quite long. For example, Gelles and Kohyama report local swelling values of 0.9% in HT-9 irradiated to 114 dpa at 420°C (i.e., the temperature for maximum swelling) in the Fast Flux Test Facility (FFTF).[7] Irradiation of nickel-doped ferritic steels in mixed spectrum reactor shows that although helium accelerates or enhances bubble nucleation and void growth, swelling remains low at all temperatures.[8]

For irradiation temperatures below about 450°C the yield stress of ferritic steels is increased by neutron irradiation, an effect primarily attributable to the dislocation loop structure resulting from the displacement damage. However, there appears to be an additional increment in strength attributable to helium.[9] Tensile properties per se do not appear to limit alloy performance, since the strength is increased and ductility remains acceptable.

The increased strength or hardening affects fracture behavior, and this is a critically important property for ferritic steels. Gelles[10] has recently reported results of irradiations of Cr-Mo steels conducted in a fast neutron spectrum (FFTF), which show that adequate toughness is maintained for damage levels as high as 100 dpa. Klueh and Alexander[11] have irradiated 9Cr-1MoVNb and 12Cr-1MoVNb steels in the mixed neutron spectrum of the High Flux Isotopes Reactor (HFIR) at 300 and 400°C to displacement damage levels up to about 40 dpa and 10 appm He. Shifts in DBTT for the 9 Cr and 12 Cr steels were 167 and 105°C at an irradiation temperature of 300°C and 204 and 242°C at an irradiation temperature of 400°C. The observation that the shift at 400°C exceeded that at 300°C indicates there is a maximum in response. Irradiation of the same alloys in a fast neutron spectrum experimental breeder reactor (EBR-II) at 390°C to 13 and 26 dpa and ~2 appm He gave shifts of 52 and 54°C, respectively,[12] indicating saturation with fluence. These results suggest that the higher helium levels produced in these alloys in the mixed spectrum reactor irradiations, which are more characteristic of fusion damage, are having a pronounced effect on the fracture behavior. Clearly, the question of irradiation effects on fracture of ferritic (martensitic) steels at fusion relevant damage levels is a question relating to the feasibility of using these alloys that must be answered. A major focus of efforts to develop ferritic steels for fusion applications must be in this area.

Conventional FM steels for elevated temperature applications, such as those described in the previous paragraphs, are based on the Fe-Cr-Mo ternary system. These alloys generally contain 2 1/4 to 12% Cr and about 1% Mo. Some alloys will also contain small amounts of carbide forming elements such as Nb and V. From the viewpoint of developing reduced activation alloys, it is necessary to remove the Mo and Nb. In recent research, new reduced activation FM steels have been developed based on the Fe-Cr-W system.[13] In these new alloys W is substituted for Mo on an atom-for-atom basis and V, Ti, and Ta are used as the carbide forming elements in place of Nb. The new alloys have mechanical and physical properties equivalent to the conventional alloys[14-17] and early indications are that *irradiation damage resistance is equal or superior to the* conventional Cr-Mo steels.[18-19]

Vanadium alloys have several properties that make them an attractive fusion reactor structural material candidate. The most common alloying elements used in vanadium base alloys (i.e., Ti, Cr, Si) have attractive activation characteristics. If not reduced by radiation damage or compatibility considerations, an upper temperature for use of vanadium alloys would be about 700°C. The combination of relatively low thermal expansion and high thermal conductivity leads to low thermal stresses at present. There is little use of vanadium alloys at high temperatures, and a very limited technological base and industrial capacity exists. In the 1960s, vanadium alloys were considered for cladding/duct applications in Liquid Metal Fast Breeder Reactor Programs and were dropped because of compatibility considerations. The United States Fusion Materials Program has undertaken the development of vanadium alloys primarily for use in a liquid lithium coolant/breeder-vanadium alloy structure system. Research has focused on corrosion and compatibility, radiation damage resistance, effects of alloy composition on mechanical properties and resistance to hydrogen embrittlement. Sensitivity of mechanical properties to the presence of carbon, oxygen, nitrogen, and hydrogen is a major impediment to the successful application of this otherwise attractive alloy system. Control of interstitial impurities is essential at all stages of production, fabrication, and in-service operation. The high solubility and diffusivity of hydrogen and hydride formation pose a number of unresolved issues relating to the containment and inventory of tritium in a vanadium alloy FWB structure. Significant progress has been made towards the development of radiation damage resistant alloys. Based on results from a systematic investigation of the effects of Cr, Ti, and Si concentrations on tensile properties, DBTT, irradiation hardening and swelling the composition V-5Ti-5Cr has been selected as a base composition for further development and optimization.[20,21]

SiC/SiC composites are attractive candidate structural materials from the viewpoint of afterheat and radioactive decay. It is presently not possible, however, to evaluate the true potential of these materials because fundamental information relating to mechanical behavior, chemical compatibility, radiation effects on mechanical and physical properties, etc., is not available. Research on these questions is now in very early stages in the world's fusion materials programs. From an engineering viewpoint, the efficacy of SiC/SiC composites as fusion structural materials will depend on technological issues such as economic fabrication of large, complex structures; suitable joining techniques; dissimilar materials transitions; etc.

3. PLASMA FACING MATERIALS — FIRST-WALL ARMOR AND DIVERTORS

The first wall structure forms the inside of the plasma chamber. The primary function of this structure is to protect the blanket structure, located directly behind it, from thermal and mechanical loads during off-normal events (plasma disruptions) and to maintain the purity of the plasma. The *divertor functions as a plasma impurity control system* to remove helium produced in the D-T fusion reaction.

The overriding considerations in selecting plasma facing materials, are maintenance of plasma purity, ability to withstand high-heat fluxes, and the ability to withstand high-energy deposition rates during plasma disruptions. Impurities are introduced into the plasma by physical and/or chemical sputtering of atoms from first-wall and divertor plate surfaces when energetic ions or neutrals from the plasma edge impinge on them. Impurities in the plasma become ionized and radiate power from the plasma as X rays. As the impurity concentration in the plasma increases, the energy gain decreases. Radiative losses increase with the atomic number of the impurity, thus materials selected for plasma facing applications generally fall into one of two categories: they are either low Z elements, such as carbon or beryllium, so that radiative losses are minimized, or high Z elements, such as molybdenum, in which case their use relies on maintaining the plasma edge temperature (or ion energy) below the sputtering threshold, thus preventing their introduction into the plasma.

The magnitude of the problem associated with heat fluxes and energy deposition rates on the first wall and divertor can be put in perspective by considering the ITER operating parameters.[22] Peak and average surface heat fluxes on the first wall are estimated to be 0.6 and 0.15 MW m^{-2} and, on the divertor, 15 to 30 and 0.6 MWm^{-2} . In a thermal quench, divertor surfaces will be subjected to energy densities of 10 to 20 MJm^{-2} in 0.1 to 3 ms. Designing first-wall and divertor structures to survive these conditions and to meet other criteria, such as temperature limits, allowable tritium inventories in the structure, etc., is a significant challenge.

It is proposed that the ITER first wall be constructed of carbon-carbon (C/C) composite tiles, which will be cooled by conduction to the actively cooled stainless steel first wall to which they are brazed. A compliant interlayer is required to prevent fracture of the tile during brazing and to improve performance under cyclic heat loads. The ITER divertor design consists of C/C composite blocks brazed to high-strength oxide dispersion-strengthened copper alloy tubes for circulation of water coolant. One feature which leads to the selection of C/C composites over bulk graphites is the ability to tailor the properties through selection of fiber type, architecture, and processing temperature. Additionally, C/C composites have high thermal conductivities, mechanical properties, and thermal shock resistance which are superior to those of the best nuclear grade graphites. Radiation effects (particularly irradiation-induced swelling) are a major limitation in graphitic materials. During neutron radiation, graphite initially undergoes densification followed by catastrophic swelling. Lifetime of the ITER C/C first wall is estimated to be only about 1 MWy/m^2. Research is now in progress on the relationship between radiation damage in C/C composites and the fabrication methods, fiber type, architecture, etc., with the goal of improving performance and extending life. Consideration is also being given to the use of tungsten or beryllium as plasma-facing materials, and refractory alloys as substrates.

4. CERAMICS

Ceramics will be used in many applications throughout a fusion reactor. Because of the uniqueness of the fusion environment the specific information needed for materials selection and system design, even for near-term fusion devices is unavailable or inadequate in many instances. Although there has been significant basic research relating composition and microstructure to properties and noted success in developing ceramics with improved mechanical and physical properties, the development of ceramics with properties tailored for fusion service has not begun. This development must be guided by the specific requirements for the various applications. In plasma-facing and high-heat flux applications, high-temperature capability and thermal shock resistance are critically important. Electrical resistivity, mechanical strength, and structural integrity are necessary in applications such as electrical insulators. Transparency to electromagnetic radiation, thermal conductivity, and fracture resistance are critical properties in the use of ceramics as windows and feedthroughs in RF heating systems. In structural applications, mechanical properties (e.g., strength, fracture toughness, creep, and fatigue) and physical properties (e.g., thermal conductivity) are important. Inside the shield, the service environment includes both displacive and ionizing irradiation, in addition to elevated temperatures, stresses, and compatibility demands. In ceramics, as with metals, many of the physical and mechanical properties are altered by irradiation. One important difference between metals and ceramics, however, is that, in ceramics, the electrical and some physical properties are affected by ionization damage produced by both gamma and neutron irradiation as well as displacement damage, which is produced predominately by neutrons. Transport properties such as electrical conductivity and loss tangent are dependent on the instantaneous concentration and mobility of carriers and are thus dependent on the instantaneous damage production rate (i.e., the gamma and neutron fluxes). These properties must be measured during irradiation to obtain data for design. A second important factor, and one which is almost totally unexplored, is the effect of very large amounts of hydrogen and helium produced by (n,p) and (n,α) reactions. With the high-energy neutrons of the fusion spectrum, the cross sections for these reactions are significantly larger for light elements (e.g., C, Si, N, O, Al), the constituents of many ceramics, than for heavier elements, the constituents of structural alloys.

5. SUPERCONDUCTING MAGNETS

Realization of fusion as an economic energy source will require development and demonstration of large, highly reliable, high-field superconducting magnets to produce the torrodial fields required for plasma confinement. The 16 TF magnets for ITER must produce a peak field of about 12 T in order to produce a field at the plasma center of about 5.3 T. ITER TF coils will most likely utilize Nb_3Sn superconductor, copper stabilizer, a glass fiber- reinforced plastic composite insulator, and 316LN stainless steel for the magnetic case. ITER magnet design criteria include radiation limits of 5×10^9 rad to the insulator and 1×10^{23} n/cm^2 (E >0.1 MeV) peak fast-neutron fluence to the conductor and stabilizer. The 316LN case structure of the magnet is very large, up to 0.5-m thickness, in order to sustain the very large loads from the Lorentz forces on the conductor. The primary materials issues relating to the ITER magnets include manufacturing technology for the very heavy section case structure and the Nb_3Sn superconductor system as well as development of an adequate data base, including the effects of service conditions on critical properties, to support the design and ensure an acceptable level of reliability.

Superconducting magnets for ITER will, to the extent possible, utilize existing materials. Figure 5, taken from the ESECOM study,[6] shows that the cost of electricity (COE) initially decreases as the magnetic field increases, reaches a minimum and then increases. The minimum in COE results from a balance between lower costs as a result of increased power density and reduced size and increasing magnet costs due to decreasing critical current densities. Advances in technology will be required for each of the materials in the magnet system for fusion to achieve its maximum economy. If fields greater than about 20 T are required, then advanced superconductors such as NbN, Nb_3Al, $Nb_3(AlGe)$, or possibly the high-temperature ceramic superconductors must be developed. Metal matrix composites may replace pure aluminum or copper as stabilizers for improved mechanical performance. Clearly, advanced materials for conduit and magnet cases will be required if fields are to exceed those of the ITER. The most important manufacturing characteristics of materials for conduit and magnet case applications are fabricability, heat treatment requirements, and weldability. Yield strength and fracture toughness are the most critical mechanical properties for superconducting magnet case alloys. Although there are metallurgical approaches to optimize the toughness or yield stress, it is a general trend that, as the yield strength is increased by adjustments in microstructure, the fracture toughness decreases. Thus, although the conventional Fe-Ni-Cr austenitic steels have many attractive properties, new alloys will be required to move off of this trend line. The Japan Atomic Energy Research Institute (JAERI), with the Japanese steel industry, have a significant program to develop improved austenitic alloys for these applications. The goal is

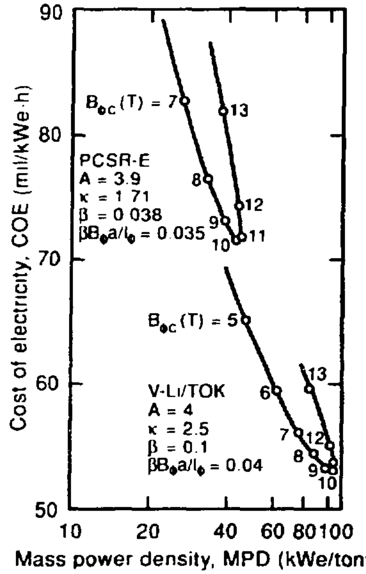

Fig. 5. Cost of Electricity as a Function of Mass Power Density for Two Fusion Reactor Concepts. The parameter, P, is the torrodial magnetic field. As the field is increased COE initially decreases as a result of higher power densities and reduced size of the fusion power core. At still higher fields COE increases because of lower critical current densities and larger magnet structures.

applications. The goal is an alloy with a yield stress of 1200 MPa and a fracture toughness exceeding 200 MPa-m$^{1/2}$. Alloys are being developed in both the Fe-Ni-Cr and Fe-Cr-Mn systems that meet these goals.[23]

SUMMARY

When all aspects are considered, fusion represents the most difficult challenge of any energy system from the viewpoint of developing the required materials. Progress over the past 15 years of research on the effects of this environment on the behavior of materials and the development of materials with properties tailored to withstand its demands gives confidence that adequate materials can eventually be developed. We now have a good understanding of irradiation damage production (displacements and transmutations) in the fusion neutron spectrum. We have explored the effects of irradiation damage on the properties of several alloy systems and have identified the most important and limiting irradiation effects. Phenomenological models have been developed for microstructural and microchemical evolution, irradiation swelling and creep, and some aspects of mechanical behavior. We have developed and demonstrated a metallurgical approach to accommodate large amounts of transmutation produced helium in structural alloys. We have identified alloy systems with attractive attributes which are also judged to have potential for development as structural alloys, a step which is critical in the systematic development of alloys for this application.

In the near future the focus of the program must remain on two points. The remaining critical issues relative to the use of type 316L stainless steel in the ITER must be addressed and assuming a favorable result a comprehensive design data base for irradiation damage levels to 30 dpa as required to support an engineering design must be generated. The efforts to develop materials for fusion power reactors must continue on a broad front for we are still not in a position to specify a reactor concept and we have no alloy or ceramic in which we have a high degree of confidence that the requirements as a structural material for economic, safe, and environmentally acceptable fusion power reactors can be met.

REFERENCES

1. J. P. Holdren, *Scientific American*, Vol. 263, September 1990, pp. 157-163.
2. *ITER Concept Definition*, Vols. 1 and 2, ITER Documentation Series, International Atomic Energy Agency, Vienna, 1989.
3. P. J. Maziasz, A. F. Rowcliffe, M. L. Grossbeck, G.E.C. Bell, E. E. Bloom, A. Hishinuma, T. Kondo, R. F. Mattas, and D. L. Smith, *Fusion Technology*, Vol. 19, May 1991, pp. 1571-79.
4. E. A. Kenik, T. Inazumi, and G. E. C. Bell, *J. Nucl. Mater.* 183 (1991) 145-53.
5. G.E.C. Bell and T. Inazumi, *J. Nucl. Mater.* 191-194 (1992) to be published.
6. *Report of the Senior Committee on Environment, Safety, and Economic Aspects of Magnetic Fusion Energy*, UCRL-53766, Lawrence Livermore National Laboratory, University of California, Livermore, CA, 1989.
7. D. S. Gelles and A. Kohyama, *Fusion Reactor Materials Semiannual Progress Report for Period Ending March 31, 1989*, p. 193.
8. P. J. Maziasz and R. L. Klueh, *Effects of Irradiation on Materials: 14th International Symposium*, ASTM-STP-1046 (N. H. Packan, R. E. Stoller, A. S. Kumas, eds.) American Society for Testing and Materials, Philadelphia, PA (1989) 35.
9. R. L. Klueh and J. M. Vitek, *J. Nucl. Mater.* 15 (1987) 272.
10. D. S. Gelles, *J. Nucl. Mater.* 149 (1987) 192.
11. R. L. Klueh and D. J. Alexander, *J. Nucl. Mater.* 179-81 (1991) 733.
12. W.-L. Hu and D. S. Gelles, *Influence of Radiation on Material Properties: 13th International Symposium* (Part II), ASTM-STP-956 (F. A. Garner, C. H. Henager, N. Igata, eds.) American Society for Testing and Materials, Philadelphia.
13. R. L. Klueh and E. E. Bloom, *Nucl. Eng. Design/Fusion*, 2 (1985) 383.
14. R. L. Klueh, *Met. Trans.* 20A (1989) 463.

15. R. L. Klueh and W. R. Corwin, *J. Mater. Eng.* **11** (1989) 169.
16. M. Tamura et al., *J. Nucl. Mater.* **141-143** (1986) 1067.
17. K. W. Tupholme, D. Dulieu, and G. J. Butterworth, *J. Nucl. Mater.* **179-181** (1991) 684.
18. D. S. Gelles, *Reduced Activation Materials for Fusion Reactors*, ASTM-STP-1047 (eds., R. L. Klueh, D. S. Gelles, M. Okada, and N. H. Packan) American Society for Testing and Materials, Philadelphia, 1990, p. 113.
19. R. L. Klueh and D. J. Alexander, *J. Nucl. Mater.*, **187** (1992) 60.
20. B. Loomis and D. L. Smith, *J. Nucl. Mater.* **191-194** (1992), to be published.
21. B. A. Loomis, D. L. Smith, and F. A. Garner, *J. Nucl. Mater.* **179-181** (1991) 771.
22. *ITER Plasma Facing Components*, ITER Documentation Series, No. 30, International Atomic Energy Agency, Vienna, 1991.
23. S. Shimamoto, H. Nakajima, K. Yoshida, E. Tada, *Advances in Cryogenic Engineering Materials*, Vol. 32, Plenum Press, p. 23.