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Materials to deliver the promise of fusion power – progress and challenges

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Abstract

High-performance reduced-activation materials are crucial for fulfillment of the promise of fusion to provide safe, economical, and environmentally acceptable energy. Three reduced activation structural materials have emerged as promising candidates, based on 8–9Cr ferritic/martensitic steels, V–Cr–Ti alloys, and SiC/SiC composites. Due to advances in understanding how to control and engineer the nanoscale phase stability required for harsh neutron irradiation environments, these reduced activation materials have unirradiated properties that are superior to commercially available analogs. Perhaps the most important accomplishment to date from fusion materials research is the radiation effects knowledge base. Models of radiation effects and supporting experiments highlight the critical role of helium production on the microstructural stability and lifetime of irradiated materials. The proposed International Fusion Materials Irradiation Facility (IFMIF) would fill a critical need for fusion materials development.

1. Introduction

Recent projections [1] indicate that the world energy consumption will double or triple within the next 50 years. Environmental issues associated with carbon dioxide emissions are causing industrialized nations to seek alternative large-scale power sources as possible replacements for fossil fuels, needed by the latter half of this century. Fusion energy offers the potential of numerous attractive features as a sustainable, broadly available, large-scale energy source, including no emissions of greenhouse gases, no risk of a severe accident, and no long-lived radioactive waste. Recent advances in the science and technology of fusion energy have dramatically improved the prospect for practical fusion power to be achieved during the first half of this century. Conversely, if fusion development is not accelerated during the next few decades, the window of opportunity for fusion to contribute as a solution to stabilization of global carbon dioxide emissions will be missed.

The current international fusion strategy is designed to demonstrate the scientific feasibility and economic and environmental attractiveness of magnetic fusion energy within ~35 years [2–4]. The European, Japanese and US fusion energy development roadmaps share many common features. In all three roadmaps the tokamak-based International Thermonuclear Experimental Reactor (ITER) for burning plasma physics and the d-Li accelerator-based International Fusion Materials Irradiation Facility (IFMIF) for fusion materials are the essential experimental facilities. The plasma physics and advanced radiation-resistant materials information obtained from ITER and IFMIF, along with initial experience with engineering blanket module tests in ITER, would set the stage for design and construction of a demonstration power plant (Demo) within about 30 years.

The worldwide fusion materials programs have a strong emphasis on structural materials R&D because reduced activation structural materials are key for reaching fusion's potential as a technologically viable energy source with no long-lived radioactive waste

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and no risk of severe accidents that would require widespread public evacuation. There are numerous other important materials systems that likewise must be successfully developed in order for fusion to become technologically viable, include high heat flux components, tritium breeding systems (including neutron multipliers, where relevant), plasma diagnostic materials and insulators, blanket coolant systems, vacuum vessel and superconducting magnet materials. The scope of the present overview is focused on structural materials.

2. Present status of fusion materials development

2.1. Overview of 25 years of progress

Considerable progress has been made towards fundamental understanding and development of advanced fusion materials in the 25 years since the 1979 Miami Beach topical meeting on fusion materials. Fusion materials research in 1979 was focused primarily on commercially available materials, with a strong emphasis on acquisition of engineering data (engineering data base) as well as fundamental radiation effects studies on candidate materials. Irradiation data were obtained in fission test reactors in order to obtain a first glimpse of radiation stability. Even at that time, there was a strong consensus on the importance of a fusion neutron irradiation facility.

In the 1990s, the need to establish a viable ITER engineering design stimulated important applied and fundamental materials R&D activities. Research topics stimulated by ITER included low temperature radiation hardening and flow localization issues (including dislocation channeling), structural joining of dissimilar metals (Cu/stainless steel, W/Cu, etc.), effects of alternative stainless steel processing methods (hot isostatic pressing, casting, etc.) on the properties and irradiation response, plasma facing materials redeposition processes and high heat flux technologies, and the mechanical properties of high-strength, high-conductivity copper alloy in their unirradiated and irradiated condition. The ITER project also stimulated considerable activities on improved engineering design criteria for structural materials, particularly for cases where neutron irradiation may produce low uniform elongations.

The selection of materials for ITER was based on a total engineering approach. This balances physical and mechanical properties, radiation effects, processing capabilities, joining, maintainability, reliability, and waste disposal considerations. A specialized formulation of austenitic stainless steel that falls within the category of Type 316L(N) was selected as the primary structural material for ITER, based on industrial capabilities to manufacture the necessary complex components and an extensive data base (unirradiated and irradiated) and design equations based on ASME and RCC-MR code requirements. Several different materials were extensively evaluated for plasma erosion protection, including beryllium for first wall and limiter, tungsten for divertor components, and carbon fiber reinforced carbon composites for the vertical target. Two types of copper alloys (oxide dispersion strengthened Cu–Al_2O_3 and CuCrZr) were selected from a wide range of initial candidate materials for first wall and divertor high heat flux heat sink substrate materials. Materials for diagnostic applications (insulators, cables, windows, fiber optics, etc.) were extensively researched and evaluated for the very challenging environment in ITER that will produce unprecedented irradiation doses on these radiation-sensitive materials in a plasma physics machine. Research is currently ongoing to determine the best material choices for a number of plasma diagnostics. The technologies

Table 1
Summary of some of the major fusion materials achievements accomplished within past 10 years [5–22]

Demonstrated via experiments and MD simulations that fusion defect production 'source term' is similar to that for fission (validates the use of fission reactors for initial testing and screening) [5]
Provisional operating windows have been established for all three classes of reduced activation structural materials, based on radiation embrittlement, thermal creep, thermal conductivity, and void swelling considerations [6–9] (fusion-relevant He/dpa condition still to be probed)
Fundamental procedures for fabricating and joining all three classes of structural materials have been established [5,10–12]
Development of miniaturized specimen test techniques: evolution from qualitative screening guide to quantitative data generation [5,13–15]
Application of Master Curve technique to unify fracture data obtained on different specimen geometries, strain rate, constraint factor, etc. [16]
Operating limits for irradiated Cu alloys defined [9,17,18]
Determined that permanent radiation-included electrical degradation is not of significant concern for ceramic insulators in next-step machines such as ITER [19,20]
Developed radiation-resistant SiC/SiC composites, based on fundamental information obtained from systematic studies on irradiated fibers, monolithic SiC, and SiC composites [21,22]

for fabricating complex, reactor-relevant scale components and the experience obtained from operations in ITER will provide a strong foundation from which to build toward Demo.

Table 1 summarizes some of the major fusion materials achievements accomplished within the past 10 years, encompassing both fundamental and applied research [5–22]. The ITER-specific accomplishments summarized in the preceding paragraphs are not included in Table 1. Numerous other technical achievements, including development of improved irradiation effects models [23], establishment of chemical compatibility limits for numerous material combinations, and technology advances applied to the IFMIF [24] have also been accomplished during the past decade.

2.2. Reduced activation structural materials

Only a handful of elements have acceptable radiological safety performance and low long term radiation levels following exposure to intense fluxes of fusion neutrons to qualify as reduced activation materials [8,25,26]. The number of options is further reduced upon consideration of the high-performance requirements for fusion: high thermal efficiency (high operating temperatures), acceptable lifetime in an intense radiation field with high thermomechanical stresses, high reliability and maintainability, and chemical compatibility with tritium breeder materials (liquid or ceramic) and associated coolants. The choice of structural material in the first wall and breeding blanket to a large degree dictates the design of the fusion reactor system. In particular, the allowable power plant temperatures, choice of coolant, and power conversion system are critically dependent on the structural material.

The international fusion materials community can take great pride in the progress in the development of three classes of reduced activation materials that have significantly superior performance compared to the high-activation candidates under investigation in 1979. These material systems are based on modified ferritic/martensitic steels (Fe–Cr–W–V–Ta), V–Cr–Ti alloys, and SiC ceramic composites. Each material system presents a set of challenges and critical issues. Neither funding nor time allows us to pursue all options through Demo evaluation. Therefore, we must build the knowledge base that lets us make selections without full system construction and testing.

Reduced activation ferritic/martensitic steels based on 8–12%Cr and 1–2%W–V–Ta solute additions share many attributes of the well-developed commercial 8–12%Cr, 1%Mo steels. The low-activation steels developed by the fusion program have mechanical and physical properties equivalent or superior to that of the commercial steels [27]. The technology for processing and joining these steels is well developed. Their good compatibility with a wide range of gaseous and liquid coolants is well established, which permits considerable flexibility in potential fusion blanket design options. These steels have shown good resistance to radiation-induced swelling and helium embrittlement in experimental tests.

The key unresolved issues for ferritic/martensitic steels include incomplete understanding of the effect of irradiation on fracture properties (particularly at low temperatures), the role of fusion-relevant helium transmutation products on the deformation and fracture of irradiated material at low and high temperatures, and possible adverse effects on plasma control and performance due to the ferromagnetic properties of the steel. Research is also being performed to understand the applicability of the fracture mechanics Master Curve approach to these reduced activation steels. Although ferritic/martensitic steels are generally reported to exhibit good resistance to void swelling, a recent reanalysis of some fast fission reactor swelling data suggests that the post-transient swelling rate may be higher than previously reported [28]. There is also evidence for enhanced cavity swelling in steel specimens irradiated to moderate doses (>40 dpa) with fission neutrons or ions when fusion-relevant levels of helium are present [29,30]. Additional controlled alloying modifications to ferritic/martensitic steels are being examined for their potential to increase the upper operating temperature limit by further improvements in thermal creep strength. Reduced activation steels such as bainitic Fe–3Cr–3W alloys [31] offer attractive properties, including possible use without tempering, and may be a competitive alternative to 8–9Cr ferritic/martensitic steels. Looking further into the future, oxide dispersion strengthened steels [32,33] may enable significant improvements in the upper temperature capability based on improved thermal creep strength.

Vanadium alloys containing Cr, Ti and Si solute along with minor amounts of C, O and N offer high performance potential due to their anticipated high operating temperature capability. These vanadium alloys have lower radioactivity levels compared to ferritic/martensitic steels, and may be suitable for advanced fusion reactor designs utilizing high wall loadings and high power density. Thermal creep studies on V–4Cr–4Ti indicate that this alloy has high creep strength for long term operation at temperatures up to $\sim$700 °C [34]. Recent work has successfully demonstrated gas tungsten arc full-penetration welds of V–Cr–Ti in controlled atmospheres without the need for post weld heat treatment [35]. Other recent research has quantified the maximum atmospheric impurity conditions to which vanadium alloys can be exposed without a protective coating. These welding and atmospheric compatibility studies have determined that very strict atmospheric control is needed whenever V alloys are exposed to temperatures above $\sim$400 °C for extended periods of time. Vanadium alloys have good compatibility with liquid lithium, and are considered to be the only viable reduced activation structural material for self-cooled lithium blanket concepts. Due to high tritium permeability and incompatibility with even low partial pressures of oxygen, self-cooled lithium blanket systems are the only breeder/coolant that is considered to be applicable for vanadium alloys.

The main feasibility issue for systems using vanadium alloys is development of insulator coatings to mitigate magnetohydrodynamic pressure drop effects in Li coolant channels. Additional research topics where further work is needed include investigations of the effect of He and displacement damage on the mechanical properties, development of new alloys containing a high density of second phase precipitates to assist in the matrix trapping of helium, investigation of possible engineering solutions to the high tritium permeability in V alloys, determination of the irradiation creep behavior at 400–700 °C, and determination of the fundamental deformation mechanisms that control high temperature thermal creep. The lack of a widespread commercial infrastructure for production of vanadium alloys is also of concern.

SiC/SiC fiber reinforced ceramic matrix composites have been investigated for potential fusion applications since the late 1980s. SiC composites offer low induced radioactivity and after heat, and are capable of operation at temperatures in excess of 1000 °C which provides the potential for very high thermodynamic efficiency in power plant systems. In addition, SiC composites can be engineered for extreme environments by tailoring of the fiber, matrix, and interphase architectures. There are numerous feasibility issues associated with this material, including uncertainties regarding the effect of neutron irradiation on the mechanical and thermal properties, the low thermal conductivity of most commercially available SiC composites, high-strength joining techniques, hermetic seals, chemical compatibility with potential liquid coolants, and the need to develop engineering structural design criteria for ceramics [10]. There are also concerns regarding high present-day fabrication costs and the limited industrial technology base for production of large-scale SiC composites.

3. Future research needs

3.1. Understanding the environment

Past experience indicates there will be unexpected surprises during the development of materials for the very challenging fusion reactor environment. New phenomena not predicted by existing theories are often discovered as we enter unexplored performance space. For example, the liquid metal breeder reactor program required a decade of fundamental experiments and modeling to understand the fundamental mechanisms controlling void swelling and irradiation creep, and more than another decade to apply this knowledge and successfully develop and qualify materials with satisfactory dimensional stability. In this case, appropriate test reactors were readily available for experimental scoping studies and model validation tests. Continual enhancements in the state-of-the-art theory and modeling efforts will minimize the number of surprises, but will not eliminate the need for experimental testing to investigate new performance space.

Overcoming radiation damage degradation is widely considered to be the rate-controlling step in fusion materials development. R&D on other areas such as joining, compatibility, and thermophysical properties are also very important, but the critical data needed to evaluate feasibility can be obtained more rapidly compared to radiation effects studies. Evaluation of fusion radiation effects requires simultaneous displacement damage and He generation. In many cases, significant effects of He on microstructural evolution are not observed until the He concentration exceeds 10–100 appm. Therefore, moderate- to high-dose irradiation studies (above $\sim$10 dpa) at fusion-relevant He/dpa values (e.g. $\sim$10 appm He/dpa for steel) are needed to investigate microstructural stability issues for fusion reactor materials. Utilizing advanced miniaturized specimen test techniques, evaluation of a suite of mechanical properties of a single material at a given temperature requires a minimum volume of $\sim$10 cm^3 with flux gradients <20%/cm. Therefore, proposed innovative small-volume 'point' neutron source concepts such as small-scale laser-based DT fusion sources would be useful for investigating microstructural stability of irradiated materials but would not replace the need for a moderate-volume intense neutron source such as IFMIF.

Table 2 compares the maximum irradiation environment and temperature conditions for structural materials in fusion and fission (first generation light water, liquid metal fast breeder, and proposed Generation IV) reactors. A more detailed comparison is given elsewhere [36]. It is worth noting that first generation fission reactors could be developed relatively quickly in the 1950s due to low operating temperatures and modest irradiation conditions. However, it still required nearly 15 years of intensive research to move from ignition (Chicago pile) to the first fission demonstration power-producing reactors. The irradiation environment for fusion is in many ways comparable to that proposed for Generation IV fission reactors that would be deployed in 15 or more years, the main difference being the much higher levels of transmutant helium that would be produced in fusion reactors.

Table 2
Comparison of the irradiation environments for structural materials in fusion and existing and proposed fission reactors

	Fission (Gen. I)	Fission (Gen. IV)	Fission liquid metal breeder	Fusion (Demo)
Structural alloy maximum temperature	<300 °C	600–1000 °C	∼600 °C	550–700 °C (1000 °C for SiC)
Max dose for core internal structures	∼1 dpa	∼30–100 dpa	∼150 dpa	∼150 dpa
Max transmutation helium concentration	∼0.1 appm	∼3–10 appm	30 (F/M steel), 75 (austenitic steel)	∼1550 appm (∼10 000 appm for SiC)

3.2. Radiation effects

With little doubt, the most important heritage of the past 25 years of fusion materials research is the knowledge base that has been generated on radiation effects. Substantial advances have been achieved in developing structural materials resistant to low-temperature embrittlement and void swelling, and fundamental knowledge has been obtained on the similarities and differences between fission and fusion neutron irradiation effects [5]. Based on sustained research efforts, the underlying philosophy has been developed for designing materials with resistance to high temperature helium embrittlement, i.e. matrix trapping of helium at uniformly dispersed nanoscale precipitates. All of these studies clearly demonstrate the complexities of the irradiated materials behavior and of the need for additional experimental data, especially data from a close simulation of the fusion environment.

A major focus for the future will certainly be the effect of helium on the microstructural stability of irradiated materials. The dominant physical phenomenon depends on irradiation temperature. At low temperatures, enhanced hardening from He bubbles and the possibility that high helium levels may cause new mechanisms for fracture toughness embrittlement are leading concerns [27,37–39]. Void swelling and phase stability issues (along with possibilities of helium modifications to irradiation creep mechanisms) are major topics at intermediate temperatures [29,30,40–43]. In particular, there is concern that void swelling may be maximized near fusion-relevant He/dpa values. It is also possible that hydrogen transmutation products may introduce synergistic effects that enhance cavity swelling. At high temperatures, the major concern is helium embrittlement of grain boundaries [44–46]. Effective management of He transmutation products in irradiated materials via matrix trapping at engineered second phases is an overriding grand challenge for successful development of fusion materials. This is particularly challenging since the formation and microstructural stability of these precipitates is strongly affected by irradiation parameters, in particular the He/dpa ratio. Specific challenges identified that need further research include the effects of transmutation-generated helium on many facets of the irradiation response and important system-dependent issues, such as the required insulators to limit magnetohydrodynamic power losses in liquid metal systems, design methods for ceramic composite structures, and the impact of ferromagnetic steels on plasma operation in magnetic confinement concepts.

3.3. Overview of irradiation facility options

Irradiation facilities capable of performing accelerated damage rate tests are essential tools in fusion materials development. The response of materials to neutron irradiation is strongly dependent on exposure temperature, displacement damage level, damage rate, and solute transmutations including H and He. For many irradiation effects phenomena, the He/dpa ratio is a useful metric for comparisons of test results obtained in different irradiation facilities. Fission test reactors are heavily used for fusion materials research. They can achieve near fusion-relevant damage rates but generally produce low He transmutants compared to fusion except in specialized cases such as Ni-containing alloys irradiated in mixed-spectrum reactors. Ion accelerators are useful for single-variable investigations of microstructural changes under accelerated damage rate conditions. High-energy proton and spallation neutron sources can generate significant quantities of helium. However, the He per unit damage is often too high, complications from other transmutation products may interfere, and the damage rate is typically lower than fusion-relevant conditions, so these facilities are expected to be of limited value to the program to develop structural materials.

Fig. 1 summarizes the helium and displacement damage regimes experimentally investigated by the US fusion materials program. Much of the work was performed in collaboration with Japanese University and JAERI researchers. The pioneering research performed in the RTNS-II 14 MeV neutron irradiation facility was instrumental in demonstrating that the displacement damage source term for D–T fusion neutrons was similar to that produced by fission neutrons, even though it was performed at He and displacement damage levels

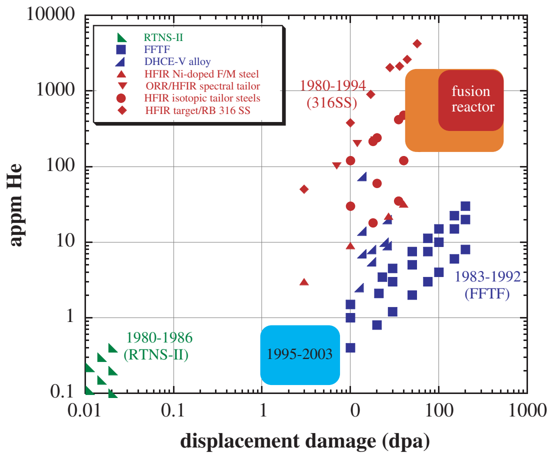

Fig. 1. Summary of helium and dose parameter range investigated by the US fusion materials program.

that were four orders of magnitude below fusion reactor relevant conditions. Research in the 1980s and early 1990s on spectrally and isotopically tailored irradiations of austenitic stainless steel in HFIR provided much of the underlying scientific knowledge and engineering data that supported the decision to use Type 316 stainless steel as the structural material for ITER. Fusion reactor relevant He levels were achieved in some of these experiments on stainless steel, and associated damage levels were about a factor of 10 below the fusion reactor levels. Irradiations in the FFTF and EBR-II fast fission reactor facilities provided substantial fundamental and applied radiation effects information on a broad range of materials at damage levels approaching fusion reactor levels, but at He/dpa levels that were typically ~50 times below fusion levels. Research in 1990–2000 on detailed mechanical property changes was of high importance for qualifying several of the materials to be used in ITER, and providing significant insight into the phenomena of fracture toughness embrittlement and loss of ductility associated with radiation hardening and dislocation channeling. Recent US/Japan collaborative research utilizing HFIR is also investigating effects of fusion-relevant He/dpa irradiations on the microstructure of ferritic/martensitic steels up to moderate doses.

Unfortunately, several of the facilities used by the fusion materials program have been shut down (RTNS-II, FFTF, EBR-II), and due to the inappropriateness of Ni-containing alloys as reduced activation materials the techniques used to generate fusion-relevant irradiation data on stainless steel mechanical property specimens cannot be replicated on current candidate fusion materials. Therefore current experimental radiation effects bulk specimen tests on promising high-performance fusion materials must necessarily be performed under conditions that are many orders of magnitude away from fusion reactor-relevant conditions. This lack of appropriate facilities highlights the urgency to construct an appropriate fusion materials irradiation facility such as IFMIF to validate fusion materials radiation effects models, uncover new phenomena, and to generate engineering data. The often suggested use of existing or planned high energy producing spallation neutron sources cannot meet this need, for the reasons enumerated earlier in this section. As discussed in Section 3.1, small-volume laser-based sources do not have sufficient constant-flux volume to satisfy physical constraint requirements for most mechanical property tests (fracture toughness, etc.).

3.4. Role of materials R&D in the path to fusion power

The complexity of a large-scale power system requires a very high reliability in the subsystems to achieve required plant availability. Many special or functional materials are of importance in addition to the structural materials. Materials in the primary circuit that are in contact with the coolant must exhibit very high chemical compatibility. The materials must be easily and reliably joined. For a fusion power system, most physical and mechanical properties of the structural materials impact technological viability, safety, and economics. The major areas of concern include thermal stress, mechanical stress, irradiation induced deformation (irradiation creep and swelling), and chemical compatibility with coolants and tritium breeding materials (requiring fundamental information on corrosion and mass transfer behavior). Within the topic of mechanical stress, several subtopics emerge which are of high importance, including plastic instability or overload conditions (determined from tensile properties of unirradiated and irradiated material over a range of test conditions), cyclic behavior (determined from fatigue and crack growth tests), time dependent deformation (creep-rupture properties), and fast fracture (determined from fracture toughness studies).

The development of new materials is a long process, involving a steady progression from basic materials science investigations to materials engineering. It begins with the identification of components, service conditions, candidate materials, and the key property requirements. For fusion, the unprecedented levels of neutron radiation are a unique and demanding consideration. Additional factors to consider are the geometries and manufacturability of components, and a system level evaluation of the compatibility of the several materials likely to be required in the system. Metrics must also be established for judging candidate and developmental materials. Table 3 shows the key steps and some of the considerations and facilities used in the specific case of the development and qualification of materials for service in a fusion power system.

Table 3
Role of materials research in the path to fusion power

Materials R&D topic	Materials facilities and activities	Plasma physics facilities and activities
Materials research to identify candidate first wall/blanket structural materials • Thermal conductivity and expansion • Activation • Mechanical properties • Chemical compatibility • Radiation damage issues • Joining and fabrication	Ion accelerators RTNS-II Fission reactors	Plasma physics confinement R&D Fusion reactor concept definition
Identify and demonstrate approaches to improve material performance Identify concept-specific issues and demonstrate proof of principle solutions, e.g. • Design with ferromagnetic material • MHD insulator for V–Li concept • Methods for design of large thermal mechanically loaded composite structures	IFMIF Fission reactors	ITER – Test Blanket Modules using reduced activation ferritic/martensitic steels will demonstrate fabrication and service in Demo relevant conditions, but at lower flux and much lower fluence levels
Development of materials with acceptable performance and demonstrate to goal life (dpa, He) Demonstrate solution to concept-specific issues on actual structural materials and prototype components Develop design database constitutive equations and models to describe all aspects of material behavior for design and licensing	IFMIF Component test facility[a] Fission reactors Confirmation and modification of performance with actual fusion environment test results	Demo conceptual designs Demo final design and construction

[a] A component test facility is proposed in some US fusion program plans. It would be designed to investigate several fusion development issues, focusing on blanket and high heat flux performance.

4. Conclusions

Fusion can only fulfill its promise of providing safe, economical, and environmentally acceptable energy if materials science and engineering can develop and deliver structural materials that meet the very challenging service requirements of a fusion power system. Considerable progress has been made towards this goal in the 25 years since the Miami Beach topical meeting on fusion materials. We now have three candidates, low activation materials systems with both promising potential and formidable challenges. Within the ferritic/martensitic steel class, analogs of commercial alloys with both low activation potential and better radiation resistance than the parent compositions have been developed. In the case of vanadium base alloys, a reference composition suitable for further alloy development has been identified. We have begun to understand the fundamental behavior and radiation damage mechanisms in SiC/SiC composites and have demonstrated control and improvement in mechanical behavior with advanced fibers and fiber–matrix interfaces. With little doubt, the most important heritage of these past 25 years is the knowledge base that we have generated on radiation effects. The status of this work clearly demonstrates the need for expanding our fundamental understanding of the materials behavior, and of the need for additional experimental data, especially data from a close simulation of the fusion environment. The proposed International Fusion Materials Irradiation Facility (IFMIF) will fill this need for materials irradiation experiments. Specific challenges identified but awaiting in-depth research include the effects of transmutation-generated helium on many facets of the irradiation response, and important system-dependent issues, such as the required insulators to limit MHD power losses in liquid metal systems, design methods for ceramic composite structures, and the feasibility of using ferromagnetic steels in magnetic confinement concepts.

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