ADVANCED MATERIALS FOR FUSION TECHNOLOGY

Steven J. Zinkle
Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831-6138 USA

Abstract

The challenging environment associated a fusion reactor (radiation, heat flux, chemical compatibility, thermo-mechanical stresses) will require the utilization of advanced materials in order to enable the successful development of fusion energy. Research supported by the international fusion materials programs and the broader materials science community is providing important advances in the development of improved materials that also satisfy the requirements for reduced long-term activation and low short-term decay heat. An overview is given regarding recent work on high-performance ferritic/martensitic and bainitic steels, nanocomposited oxide dispersion strengthened ferritic steels, vanadium alloys, and SiC composites, which are candidate structural materials for fusion systems. Several of these advanced alloys developed by fusion researchers are being spun off for near-term commercial applications in other fields such as fossil energy.

Keywords: ferritic steels, oxide dispersion strengthened steels, vanadium alloys, silicon carbide composites

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Corresponding author:
Steven J. Zinkle
Metals and Ceramics Division
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6138 USA
Tel. +1-865-576-7220
Fax +1-865-241-3650
Email: zinklesj@ornl.gov

1. Introduction

Development of new high-performance structural materials ranging from aircraft turbine components to pressure vessel materials traditionally requires sustained research and development (R&D) over a period of decades. For example, a worldwide effort involving hundreds of scientists over a period of 20 years overcame several major technical hurdles [1] to successfully develop Ni_3Al intermetallic high temperature alloys with sufficient ductility and fabricability for commercial applications in 2003. In keeping with historical industrial practice for the introduction of new materials, the initial commercial use of this advanced material is for a relatively benign, low-impact application involving furnace fixtures and rollers used for steel fabrication and heat treatment of automotive parts [2]; higher performance structural applications (particularly involving human safety) will presumably follow once sufficient industrial experience is obtained. Similarly, a large 15 year R&D effort recently developed four successfully improved generations of Si_3N_4 ceramics with a resultant four-fold increase in high-temperature strength that enabled these monolithic ceramics to become reliable structural materials for certain low-stress engineering applications [3].

The development challenges for these materials systems pale by comparison to that for fusion materials, which is arguably the greatest materials development challenge in history. The combination of high temperatures, high radiation damage levels, intense production of transmutant elements (in particular, H and He), and high thermomechanical loads that produce significant primary and secondary stresses and time dependent strains requires very high performance materials for fusion energy systems. In contrast to first-generation (late 1950s) demonstration fission reactor plants where the maximum damage level achieved by any structural material was on the order of one displacement per atom (dpa), the structural materials in the first demonstration fusion reactor will be expected to satisfactorily operate up to damage levels approaching 100 dpa or higher. The relatively high production rate of helium and hydrogen transmutation products generated in materials by fusion neutron absorption will also require special mitigation techniques to minimize cavity swelling and embrittlement (highly stable second phase microstructures) that were not needed for materials in first- and second-generation fission reactor systems.

At low temperatures, defect cluster accumulation in the matrix can produce high hardening with accompanying embrittlement and the localized deformation phenomenon known as dislocation channeling may create high stress concentrations at grain boundaries. At high temperatures, formation of helium cavities at grain boundaries can lead to severe intergranular embrittlement. Therefore, the matrix regions and grain boundaries must be designed to mitigate the severe neutron radiation effects. Finally the unique requirement to restrict alloying compositions to so-called "reduced activation" elements in order for fusion to achieve its environmental "low-impact" potential produces considerable constraint on the number of options for development of improved materials.

In order to meet these daunting challenges, a suite of advanced materials with impressive performance has been developed by the international fusion materials community [4-6]. It is worth noting that the concept of reduced activation materials for fusion was introduced approximately 20 years ago [7], and therefore none of the reduced activation structural materials currently under investigation (Fe-9%Cr ferritic/martensitic steels, V-Cr-Ti alloys and SiC/SiC ceramic composites) existed 15 years ago in their current compositions. These materials developments have been guided by materials science investigations to determine the underlying physical phenomena that control the intrinsic behavior of body centered cubic metals (applicable for Fe-9%Cr ferritic/martensitic steels and V alloys) and ceramic composites (applicable for SiC/SiC composites). Advances in miniaturized specimen test techniques have enabled engineering-relevant data to be obtained at lower cost in significantly smaller volumes. For example, the proposed international fusion materials irradiation facility (IFMIF) [5] high flux test volume is 1/20th that considered to be the minimum required test volume 15 years ago [8], with a resultant substantial reduction in facility cost. Improvements in the knowledge base regarding fundamental physical mechanisms of deformation and fracture have led to the development of provisional multiscale models on generic radiation-resistant microstructures that can be experimentally tested today in fission reactor irradiations and using IFMIF in the future.

This paper briefly reviews the advanced reduced-activation materials currently under investigation and summarizes several new classes of materials that may offer potentially even greater performance in the future. The focus will be placed on a variety of advanced steels, with some discussion also provided on vanadium alloys and SiC/SiC ceramic composites.

2. Ferritic Steels

The international adoption of the mandate that structural materials for fusion should not produce high levels of long-lived radioactive products and that short lived products should not produce unacceptable safety consequences (due to high decay heat and/or volatile species that could be released in the event of loss of coolant to the blanket region) produces severe limitations on the alloying elements that can be used to fabricate high performance steels [9,10]. Conventional alloying additions such as Nb, Mo, Co and Ni are not permitted in these so-called reduced-activation steels. Nevertheless, a series of reduced activation 8-9%Cr ferritic/martensitic steels with superior mechanical properties and improved radiation resistance compared to commercial steels such as HT-9 (Fe-12%Cr steel) or T91 (Fe-9%Cr-1%Mo) have been formulated based on evolving knowledge of radiation-resistant microstructures. The metallurgical reasons why steels with 7-9%Cr received the greatest attention in the worldwide fusion reduced activation steel R&D program have recently been reviewed by Klueh [11]. Briefly, 12%Cr steels often contain the δ-ferrite phase that lowers the fracture toughness. If carbon or manganese is added to the steel to suppress δ-ferrite, either $M_{23}C_6$ precipitates are formed (which tend to reduce the fracture toughness) or the chi phase is formed during irradiation, which has been linked to embrittlement [12].

The compositions of the 8-9%Cr steels under current investigation in the international fusion program are summarized in Table 1. The main compositional differences between these steels include tungsten contents of 1 to 2% (lower level in EUROFER in order to improve the tritium breeding capability for a ceramic breeder blanket) and a factor of six variation in Si content (0.05 to 0.3%). The heat treatment for all of these steels consists of a normalizing exposure at ~1000 to 1100˚C for ~0.5 h followed by a tempering heat treatment at 740 to 780˚C for ~2 h. Cooling after normalizing produces the high strength but brittle martensite structure. The subsequent tempering treatment is crucial for obtaining an optimized balance between strength and fracture toughness. Short term (>1 h) heat treatment at temperatures above 750˚C or prolonged exposure (>1000 h) at temperatures above 650˚C can lead to overtempering of the martensite structure and accompanying softening of the steel.

As reviewed elsewhere [4,6,13-18], the reduced-activation 8-9%Cr ferritic/martensitic steels developed by the international fusion materials community have unirradiated thermal creep strengths and fracture toughness properties that are equal or superior to commercially available 9Cr-1Mo steels. Of even greater importance is the observation that the fusion reduced activation steels have significantly better resistance to low temperature radiation embrittlement compared to commercial steels [6,15,19]. This offers the possibility of significantly greater flexibility in potential operating temperatures for the reduced activation steels in fusion structural applications.

Several different options are being pursued for development of reduced activation steels with improved capabilities compared to current-generation reduced-activation steels. Incremental modifications in solute (precipitate) composition and minimization of microstructural defects offer the potential to increase the upper operating temperature limit of Fe-8-9%Cr martensitic steels by ~50-100˚C [11,20]. Alternatively, high strength 3%Cr steels with a bainitic structure [11,21] may enable lower-cost, high-performance steels, which would be of particular importance for large components such as the vacuum vessel. Two 50 ton heats of reduced activation bainitic steel Fe-3Cr-3W-0.25V-0.1C with and without 0.07%Ta have recently been fabricated in the US and are currently undergoing mechanical testing to provide the database for an ASME code case [11]. Tensile and short-term (up to 5000 h) thermal creep tests performed to date have found the 3Cr steel strengths are comparable or superior to existing 8-9Cr steels. Figure 1 summarizes some thermal creep data for the new 3 Cr steel and commercial 2 1/4 and 9Cr steels [11]. Neutron irradiation testing is needed to determine if these 3Cr steels have good resistance to radiation damage as was previously found for 2 1/4 Cr reduced activation steels. The key for the improved performance in both the 8-9Cr and 3Cr steels is development of a high density of nanoscale precipitates.

Recently developed nano-composited dispersion-strengthened ferritic/martensitic steels containing 9-14%Cr offer the highest operating temperature capability of steels.

These revolutionary steels differ from conventional oxide dispersion strengthened (ODS) steels in that the particles are smaller (~2 nm radius), of higher density, and of more uniform distribution [22-25]. Another important factor is that the particles in the new dispersion-strengthened steels are not simple oxides, but instead are a hybrid precipitate/oxide mixture [24]. There are two main options for ODS steels, based on the pioneering work by Ukai and coworkers [22,23]. Ferritic ODS steels (typically containing 12-16%Cr) have better high temperature oxidation resistance and thermal creep strength and hence offer the highest temperature capability (temperatures up to 800˚C may be achievable). The key disadvantages of the 12-16Cr ferritic ODS steels are anisotropic mechanical properties and relatively low fracture toughness particularly for crack propagation along the extrusion direction. Martensitic ODS steels containing ~9%Cr exhibit nearly isotropic mechanical properties after heat treatment and better fracture toughness. The martensitic ODS steels are limited to maximum temperatures of 650-700˚C and have marginal oxidation resistance at high temperatures.

The key R&D issues for ferritic/martensitic steels include 1) verify that ferromagnetic structures are acceptable for magnetically confined reactors, 2) expand the low-temperature operating limit by developing alloys with improved resistance to low-temperature (<350˚C) embrittlement, 3) expand the high-temperature operating limit by developing alloys with improved resistance to thermal creep and high-temperature helium embrittlement, 4) examine the effects of fusion relevant helium generation rates on the structural stability (void swelling behavior, etc.) and 5) resolve system-specific compatibility issues (e.g., tritium barrier development, effect of magnetic fields on Pb-Li corrosion behavior, etc.). In order to fully capitalize on the improved performance capabilities of ODS steels, additional R&D is also needed on joining techniques (e.g., friction stir welding), improvement in the uniformity of properties, and investigation of the long-term stability of the nanoscale particles during neutron irradiation.

3. Refractory alloys and SiC/SiC composites

Alternative advanced reduced activation structural materials such as vanadium alloys and SiC ceramic composites are under investigation by the worldwide fusion materials programs [6,26]. These material systems offer the potential for higher operating temperatures and therefore improved thermodynamic efficiency compared to steels, but are not as well-developed commercially and require considerable additional research to investigate engineering feasibility. Scoping research is also being performed on chromium [27], molybdenum [28-31] and tungsten [32] alloys as potential fusion structural materials.

For vanadium alloys, most activity is focused on V-4%Cr-4%Ti which offers a good compromise between strength and fabricability. Recent thermal creep testing has shown this alloy has adequate long-term strength for temperatures up to 700-800˚C [33], depending on the design. As shown in Fig. 2, it is noteworthy that V-4Cr-4Ti has nearly twice the allowable design stress of the space-reactor refractory alloy Nb-1Zr at intermediate temperatures (~125 vs. 70 MPa) [34], and the maximum temperature capability of V-4Cr-4Ti is much higher than Nb-1Zr on a homologous temperature basis (0.49 T_M vs. 0.41 T_M for a design stress of 50 MPa, where T_M is the melting temperature). One of the major issues for V-Li systems still to be resolved is the development of an appropriate electrical insulating coating to minimize pressure drop and heat transfer inequalities in coolant channels associated with magnetohydrodynamic effects for Li flowing across magnetic field lines [35]. Numerous additional issues need further investigation, including high temperature He embrittlement, void swelling and phase stability under fusion-relevant irradiation conditions, the effect of impurities on corrosion processes, and the upper temperature limit for V alloys in contact with flowing Li during long-term exposures.

Silicon carbide composites are unique engineered materials based on woven SiC fibers (~10 μm diameter), infiltrated SiC matrix, and a thin compliant layer between the fibers and matrix (interphase) that is designed to impede propagating cracks and to allow controlled deformation [36,37]. Silicon carbide composites offer the greatest potential for very high temperature operation among the candidate reduced activation fusion structural materials, but require considerable research and development to resolve engineering feasibility and manufacturing issues. Issues receiving greatest attention include new fabrication methods that may further improve performance and lower the fabrication cost [38,39], investigation of fundamental radiation effects including thermal conductivity degradation, void swelling, irradiation creep and high temperature helium embrittlement, and development of structural design rules for ceramic composites. Due to the development of improved SiC composites, experimental studies are now increasing the emphasis on acquisition of engineering data (e.g., uniaxial tensile properties) instead of simple qualitative screening tests such as flexural bend strength. Figure 3 compares the tensile test behavior of silicon carbide composites containing three different types of interphases following neutron irradiation to 1 dpa at 800˚C [40]. The composite containing multilayer SiC interphase exhibited the best irradiated behavior, whereas pseudo porous SiC interphase with embedded islands of glassy carbon had the lowest irradiated strength.

Research on the physical mechanisms responsible for degradation of first generation commercial SiC/SiC composites have led to dramatic improvements in radiation resistance and thermophysical properties. Whereas commercially available SiC composites tested in the early 1990s suffered large decreases in strength after neutron exposures equivalent to a few days operation in a fusion reactor [41], recent 3rd generation SiC/SiC composites have shown no degradation after irradiation at an order of magnitude higher dose (~10 dpa). These encouraging results might enable introduction of SiC composites in low-risk applications in ITER test blanket modules. As discussed elsewhere [37,39], numerous feasibility issues need to be resolved before SiC/SiC composites would be ready for fusion structural applications.

4. Near-term commercial applications

In general, the microstructural modifications pursued to enhance the radiation resistance of fusion materials also improve the overall unirradiated properties. This can lead to the development of improved materials for near-term commercial applications. For example, a high density of finely dispersed precipitates that are resistant to coarsening or dissolution is a key feature in high-performance structural materials for both irradiation stability and thermal creep strength. Similarly, the improved stoichiometric SiC fibers and tailored interphases that are essential for radiation-resistant SiC/SiC composites also provide improved elastic modulus matching and load transfer balancing between the fibers and matrix that leads to high mechanical performance for nonirradiation applications.

The observations of phase evolution that occur in stainless steels during elevated temperature irradiation and the thermodynamics derived from these observations have provided the foundation for development of several stainless steels (both castable and wrought) with dramatically improved high temperature mechanical properties. Several of these alloys are now used in commercial applications. Figure 4 compares the microstructures of cast austenitic stainless steel processed by standard and improved techniques [42]. The new cast steel has very high thermal creep resistance (e.g., 23000 h creep rupture time at 850˚C for an applied stress of 35 MPa, compared to 500 h for standard material). Precise balancing of a wide number of solute additions was required in order to produce the desired microstructure. The improved creep resistance is due to the formation of a fine dispersion of stable nanoscale MC precipitates, and it has good resistance to creep cavitation and embrittling grain boundary phases such as sigma or Laves due to the controlled introduction of specific solute atoms (B, C, P) that retard formation of these phases. The improved steel was developed by utilizing reactive solute such as Ti, V and Nb, which are known to enhance MC precipitate formation during irradiation, and taking advantage of catalytic effects (i.e., Si enhances Fe_2Mo or M_6C formation during irradiation). Selective additions of Nb were used to replace some of the Ti in order to promote formation of fine-scale NbC (globular TiN preferentially formed instead of TiC in the original alloy). Other specially tailored wrought austenitic steels have been developed for microturbine recuperators and other high temperature applications [43,44]. As another example, the austenitic steel DIN 1.4550 was replaced by DIN 1.4541 in the vessel and inner plating of commercial pressurized water reactors (Nb solute replaced by Ti in the steel). This innovation reduced the decay time for hands-on maintenance for decommissioning from 30000 years to 20 years.

As discussed in section 2, a 3Cr bainitic steel developed by fusion is under active technology transfer to industry due to its dramatically improved tensile and thermal creep strength compared to existing 2 1/4Cr steels. It is particularly attractive because the joining of large sections may not require a post weld heat treatment. The 9Cr reduced-activation ferritic/martensitic steel developed by fusion has led to the development in Japan of 9Cr-3W strengthened by nanoscale MX nitrides for the boilers and turbines in ultra supercritical steam fossil energy power plants [20]. The new steel (9Cr-3W-3Co-VNb-0.05N-0.002C) has a factor of 100 longer creep life at 650˚C ($t_R=10^4$ h) for an applied stress of 140 MPa compared to conventional P92 steel (9Cr-0.05Mo-1.8W-VNb).

5. Role of modeling and experimental validation in fusion materials development

Development of materials for the harsh fusion environment requires a firm understanding of the underlying physical phenomena controlling their performance. Such a science-based approach shortens the time to develop fusion materials, and can also be harnessed to develop a range of improved high temperature materials for non-fusion applications. A comprehensive theory and modeling program that is well integrated with experimental studies on existing materials science facilities (corrosion loops, fission test reactors, ion accelerators, etc.) is very valuable for accelerating the development of fusion materials. Furthermore, the recent rapid progress in computational materials science (in conjunction with experimental validation tests in existing facilities) may lead to resolution of several of the scientific questions that are critical for successful development of fusion materials. However, an enhanced theory and modeling program does not replace the need for a dedicated neutron source such as the International Fusion Materials Irradiation Facility, IFMIF to fully develop and qualify materials for a demonstration fusion reactor [45]. Experimental validation of the behavior of materials under near-prototypic fusion reactor conditions is necessary for confirmation of model predictions and to gain approval from licensing and capital investment authorities. As noted in the SOFT-23 opening plenary presentation [46] and in various international fusion energy development roadmaps [26,47,48], construction of a fusion materials neutron irradiation facility is needed in parallel with ITER in order for fusion energy to potentially become a major worldwide energy source by ~2050.

6. Conclusions

Development of structural materials for demanding environments such as fusion is a long-term endeavor, historically requiring one or more decades of research in order to make incremental improvement. Acceleration of this schedule or development of materials for radically harsher environments than existing experience must be based on utilization of advanced materials science principles. The daunting materials challenges posed by fusion energy (high neutron irradiation fluxes and transmutant helium levels, high operating temperatures and thermal fluxes) are being addressed by a three-pronged reduced activation materials approach based on ferritic/martensitic steels, refractory alloys, and SiC/SiC ceramic composites. For the metallic alloys, the major focus is on creation of highly stable, finely dispersed nanometer-scale precipitates or solute clusters. It is noteworthy that such precipitate or dispersion hardening generally improves the thermal creep strength in addition to providing enhanced radiation resistance, thereby providing a pathway for near-term commercial utilization of advanced new materials.

Several options exist for advanced ferritic steels. Evolutionary ingot-based precipitate- or dispersion-strengthened steels (bainitic, tempered martensitic, or ferritic structures) provide incremental performance improvements on alloy systems with generally good radiation resistance. Oxide dispersion strengthened ferritic/martensitic or ferritic steels offer the potential for revolutionary improvements in high temperature performance and radiation resistance, but require advances in joining technology as well as long-term unirradiated and irradiated testing to examine their behavior.

Vanadium alloys offer attractive high temperature capability from a conventional ingot metallurgy process and thus may be an attractive alternative to ODS ferritic steel options. High temperature helium embrittlement and development of insulator coatings to minimize magnetohydrodynamic pumping pressure variations are the major R&D issues.

SiC/SiC composite R&D has progressed from initial qualitative screening studies to measurement of engineering-relevant mechanical properties (unirradiated and irradiated). Although numerous feasibility issues remain to be resolved before SiC composites can be utilized as structural materials in fusion systems, the rapid pace of advance may allow SiC/SiC composites to be tested in nonstructural roles in ITER test blanket modules.

In concert with enhanced theory and modeling activities and continued utilization of existing experimental facilities, an intense neutron source such as IFMIF is needed to develop and qualify fusion structural materials for expeditious development of fusion energy.

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Fig. 1. Creep rupture curves at 600˚C for new 3Cr steels and three commercial 2 1/4Cr or 9Cr steels. Note that the creep strength of 3Cr-3WVTa steel is similar for both the normalized and normalized-and-tempered conditions [11].

Fig. 2. Comparison of the stress-temperature design window for Nb-1Zr and V-4Cr-4Ti.

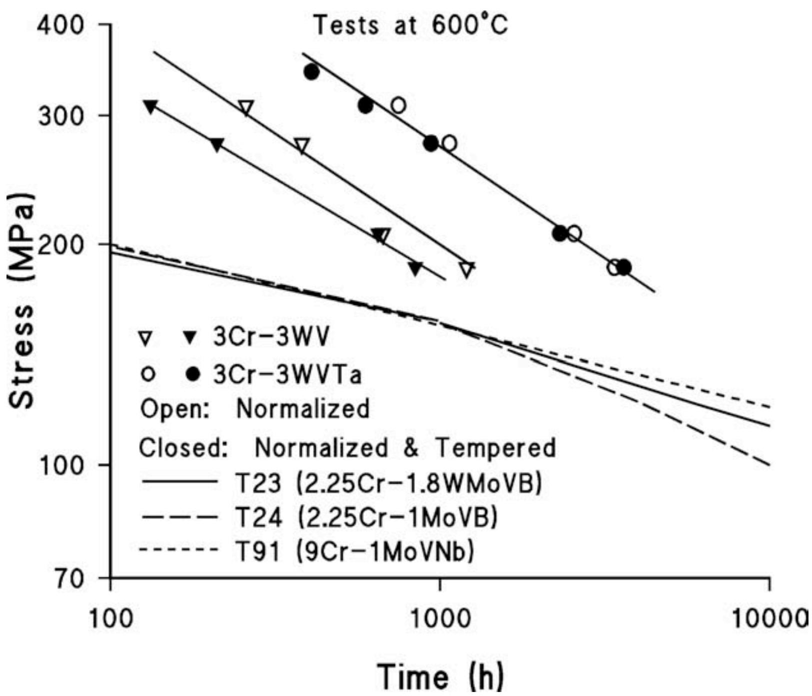
Fig. 3. Comparison of the tensile stress vs. strain curves for SiC composites (Hi-Nicalon Type S fibers with chemical vapor infiltrated SiC matrix) containing interphases of multilayer SiC (ML), pyrolitic carbon (PyC) or pseudo porous SiC (ppSiC) following neutron irradiation at 800˚C to 1 dpa [40].

Fig. 4. Comparison of the as-cast microstructures of (a) conventional and (b) improved CF8C cast austenitic stainless steels [42].

List of Tables

Table 1. Nominal composition of reduced-activation ferritic-martensitic steels (wt.%, balance Fe).

Steel	Cr	W	V	Ta	Mn	Si	C	N	B
9Cr2WVTa	9.0	2.0	0.25	0.07	0.4	0.30	0.10	--	--
F82H	8.0	2.0	0.2	0.04	0.5	0.2	0.10	<0.01	0.003
JLF-1	9.0	2.0	0.20	0.07	0.45	0.08	0.10	0.05	--
EUROFER97	8.5	1.1	0.2	0.1	0.4	0.05	0.12	0.02	<0.001



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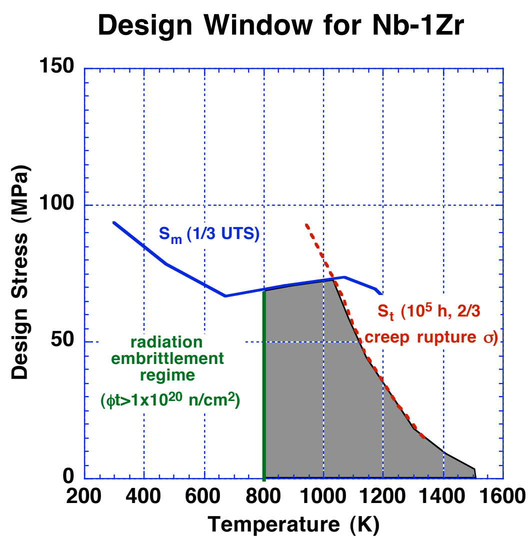

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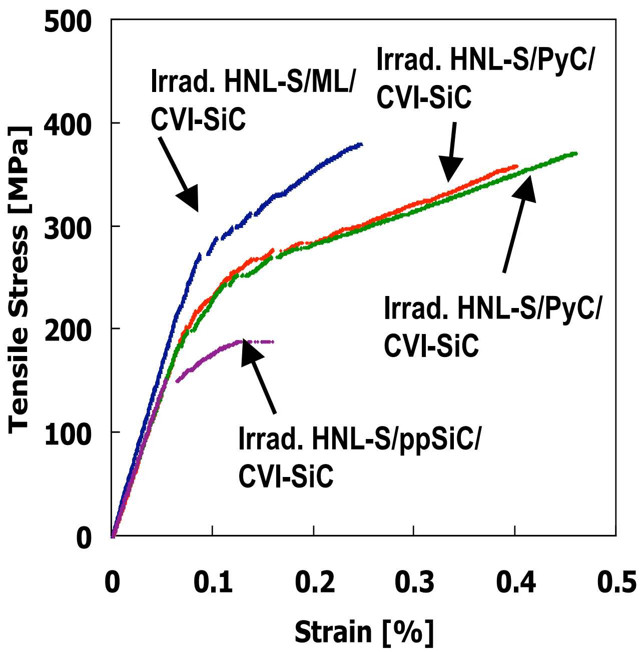

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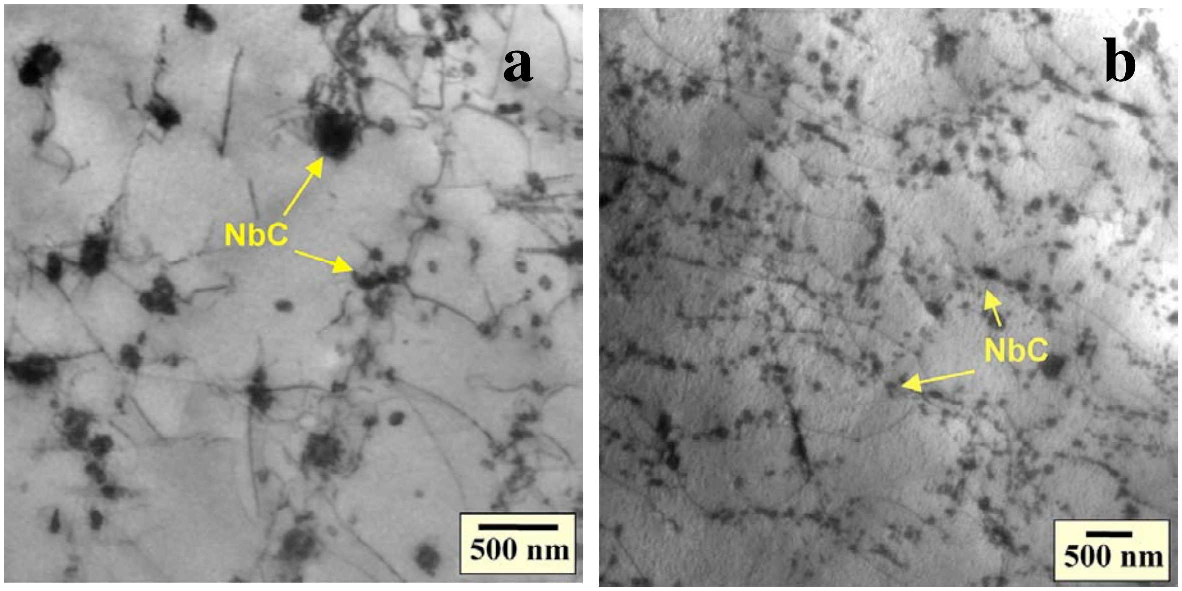

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