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PAPER TITLE (upper and lower case)

Structural Materials for Fusion Reactor Blanket Systems

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THE BLANKET is one of the largest and most complex systems in a fusion power reactor. In a D-T fueled reactor the blanket system recovers the energy produced by the fusion reaction and provides for breeding and recovery of tritium for the fuel cycle. The blanket must operate in a severe neutron radiation, thermal, chemical, mechanical, and electromagnetic environment.

Consideration of the required functions of the blanket and the general chemical, mechanical, and physical properties of candidate tritium breeding materials, coolants, structural materials, etc., leads to acceptable or compatible combinations of materials. The presently favored candidate structural materials are the austenitic stainless steels, martensitic steels, and vanadium alloys. The characteristics of these alloy systems which limit their application and potential performance as well as approaches to alloy development aimed at improving performance (temperature capability and lifetime) will be described. Progress towards understanding and improving the performance of structural materials has been substantial. It is possible to develop materials with acceptable properties for fusion applications.

FUNCTIONS AND ENVIRONMENT OF A FUSION REACTOR BLANKET

The fusion reactor blanket is one of the largest and most complex systems in a fusion reactor. Materials in the blanket are subjected to an extremely demanding environment in which neutron irradiation, chemical compatibility, mechanical loads creating primary stresses, cyclic heat fluxes creating significant thermal stresses, and elevated temperatures all contribute.

This paper will briefly review the functions of the fusion reactor blanket, the possible choices for materials other than the structural material and the environment to which the structural material will be subjected. The major issues which dictate selection of the structural material are thus identified.

No single alloy or alloy system is satisfactory for all proposed design concepts. The attractive as well as the limiting characteristics of several alloy systems for fusion reactor first wall and blanket structural applications will be summarized and the leading contenders identified. Examples of the substantial progress made in developing alloys with properties tailored for the fusion environment will be given.

Figure 1 presents a schematic of a fusion reactor blanket system. Progressing outward from the plasma is the first wall which may be coated with a material of low atomic number to lessen plasma-wall interaction problems (see paper by L. K. Wilson in these proceedings). Next is a neutron multiplier to enhance tritium breeding in the breeding portion of the blanket. The neutron multiplier may be incorporated in the breeder material (e.g., 17 Li-83 Pb). If the lithium atom density in the breeding portion of the blanket is sufficiently high (e.g., liquid lithium breeder) sufficient tritium breeding can possibly be attained without a multiplier. Progressing outwards is a structure which will contain the tritium breeder, lithium or a lithium containing compound, followed by a reflector to increase the neutron flux in the breeding region and finally a shield to reduce radiation damage to the magnets. For some designs liquid lithium will function as both breeder and coolant. Other concepts require a separate coolant to remove heat and possibly a separate tritium recovery fluid.

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Components of a Fusion Reactor Blanket System

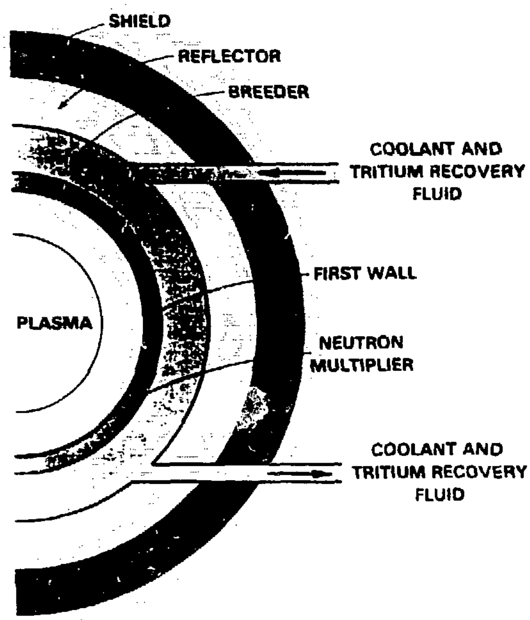

Fig. 1 — Schematic of a fusion reactor blanket. Candidate materials for the coolant, tritium breeder, neutron multiplier, and reflector are identified in Table 1.

Candidate materials for tritium breeder, coolant and neutron multiplier are listed in Table 1. Smith has reviewed the characteristics, advantages and disadvantages of these materials.[1] The liquid metal breeders offer better tritium breeding potential than the ceramic compounds and may serve as both breeder and coolant. The solid breeders will clearly need a neutron multiplier (with the possible exception of Li_2O) and a separate coolant and circulating tritium recovery fluid. A liquid metal as breeder and coolant offers a simpler design, a relatively low operating pressure, the possibility of high operating temperatures and good tritium recovery. The solid breeders offer some safety advantages because of their lower reactivity with water and the environment. However, tritium recovery and containment presents problems for the solid breeders. Magnetohydrodynamic effects pose serious constraints for the liquid metal cooled systems. Pressurized water as a coolant has excellent heat transfer properties. On the negative side water requires a high operating pressure, is reactive with some of the breeders, presents difficulties with tritium recovery and offers limited thermodynamic efficiency. Helium as a coolant is chemically nonreactive and offers easy cleanup in the event of a leak. However, helium requires high operating pressures and temperatures and this directly effects selection of a structural material. Helium also requires significant pumping power and is difficult to contain. Molten salts offer the possibility of low pressure operation and relatively easy tritium recovery. Concerns include relatively high melting temperature which complicates start up, residual radioactivity and stability of the salt.

Table 1 — Candidate Materials for Breeder, Coolant, and Neutron Multiplier

Tritium Breeder	Coolants	Neutron Multiplier
Liquid Metals		
	H_2O	Be
	Li	Pb
Li	17 Li—83 Pb	
17 Li—83 Pb	He	
	HTS	
Ceramics		
Li_2O		
Li_8ZrO_6		
$LiAlO_2$		
Li_2SiO_3		
Li_2ZrO_3		

The primary issues relative to the structural material which derive from physical and chemical properties of the breeder and coolant are (1) compatibility, (2) tritium containment, (3) required operating temperatures, and (4) pressure loads which result in primary stresses.

A second major issue relative to the structural material is radiation damage. The 14 MeV neutrons produced in the plasma deposit their energy in the first wall and blanket. In doing so they create damage in the metal lattice.

The two primary forms of radiation damage are (1) displacement damage, in which atoms are knocked from their normal lattice sites thus producing vacant lattice sites (vacancies) and atoms in interstitial positions (interstitials) and (2) transmutations. In a typical fusion neutron spectrum the fractional atomic displacement rate is 0.35×10^{-6} (dpa/s)/(MW/m^2). At a wall loading of 5 MW/m^2 each atom in the candidate structural materials will be displaced from its lattice site about 55 times a year.

Transmutations change the alloy composition. Of particular importance is helium which is produced from (n,α) reactions. Helium is highly insoluble in the metal lattice and can drastically affect microstructure and properties.

Table 2 compares the amount of damage produced in fast reactor structural alloys (i.e., fuel cladding and duct) to the damage produced in candidate structural alloys in a

Table 2 — Damage Production in Typical Reactor Components

Alloy	Residence Time or Exposure	Damage Production dpa	He at. ppm	H at. ppm
Fast Reactor Fuel Cladding and Duct				
Austenitic stainless steel	3 years[a]	150	75	1350
Martensitic steel		150	30	450
V-15Cr-5Ti		150	12	90
Fusion Reactor First Wall				
Austenitic stainless steel	15 MWy/m^2[b]	165	2205	7980
Martensitic steel	15 MWy/m^2	165	1650	6750
V-15Cr-5Ti	15 MWy/m^2	165	705	3675

[a]Typical of EBR-II or FFTF (equivalent fluence 3×10^{27} cm^{-2}).
[b]Typical neutron fluence for three year operation.

fusion spectrum. The displacement damage production rate is similar for the two radiation environments. The helium generation rate in the high energy fusion spectrum is much higher for all alloys but does vary considerably between the various alloy systems.

Only a small fraction of the vacancies and interstitials produced during irradiation survive. Most are annihilated at sinks. However, some vacancies and interstitials cluster with like defects, forming two-dimensional platelets whose perimeters are dislocation loops. Vacancies may also cluster to form three-dimensional clusters, cavities, or voids. Irradiation temperature is the most important variable in determining how this structure evolves with continuing damage production and thus in determining which physical and mechanical properties will be altered.

The evolution of the microstructure and microchemistry is a strong function of the irradiation temperature, as illustrated in Fig. 2. These results are for an austenitic stainless steel having a face-centered cubic structure. The evolution of the structure is similar in alloys with a body centered cubic structure, although the magnitude and exact temperature dependence may differ. In a temperature range in which a given type of defect is formed, the concentration of that defect generally decreases and the size increases with increasing temperature. Below about 0.35 T_m (T_m is the melting temperature in degrees Kelvin), the radiation-produced vacancies are essentially immobile on the time and distance scale of relevance while the interstitials remain highly mobile. Most interstitials diffuse through the lattice, eventually recombining with vacancies. A small fraction survive, however, and precipitate in the form of interstitial dislocation loops (Fig. 2b). Above 0.35 T_m, vacancies become increasingly mobile. In the range 0.35—0.55 T_m, a structure consisting of cavities, which are essentially voidlike (empty), and a dislocation network evolves (Fig. 2c). Over the 0.35—0.55 T_m range, various precipitates and segregated regions also form (Fig. 2c). Some of these precipitate phases also form in unirradiated material, but irradiation induces phases not normally appearing in the composition range and often changes the density, size distribution, and composition of the phases that normally would form under equivalent thermal treatment. At higher temperatures, equilibrium bubbles, i.e., cavities where the gas pressure supports the surface tension, at grain boundaries and other structural features are the only stable irradiation-produced defect (Fig. 2d).

The gradient in radiation production and volumetric heat generation with distance from the first wall is an important aspect of the fusion environment. Figure 3 shows the volumetric heat generation in lithium and radiation damage production in dpa/y in the structural material for a neutron wall loading of 5 MW/m^2. These values are for a lithium cooled design; the trend is the same but the effect is even more dramatic in other blanket concepts. Damage production decreases from 50 to 5 dpa/y in the first 0.25 m. This suggests that materials with capability to withstand higher heat fluxes and possibly higher radiation damage levels might be used in regions nearest the first wall while in regions removed from the first wall materials with better overall engineering characteristics (fabricability, weldability, compatibility, cost, etc.) could be used.

Compatibility with the breeder material and/or coolant is an important factor in the selection of the structural material and, in some cases, in establishing the operating temperature limits of the structure. For example, copper and aluminum are not compatible with lithium as the breeder or coolant. The maximum operating temperature in a Li-Pb stainless steel system will be severely limited by compatibility considerations. Compatibility concerns include both corrosion/mass transfer effects in the liquid metal systems and possible effects on the mechanical integrity of the structure, e.g., stress-corrosion of austenitic steels in pressurized water.

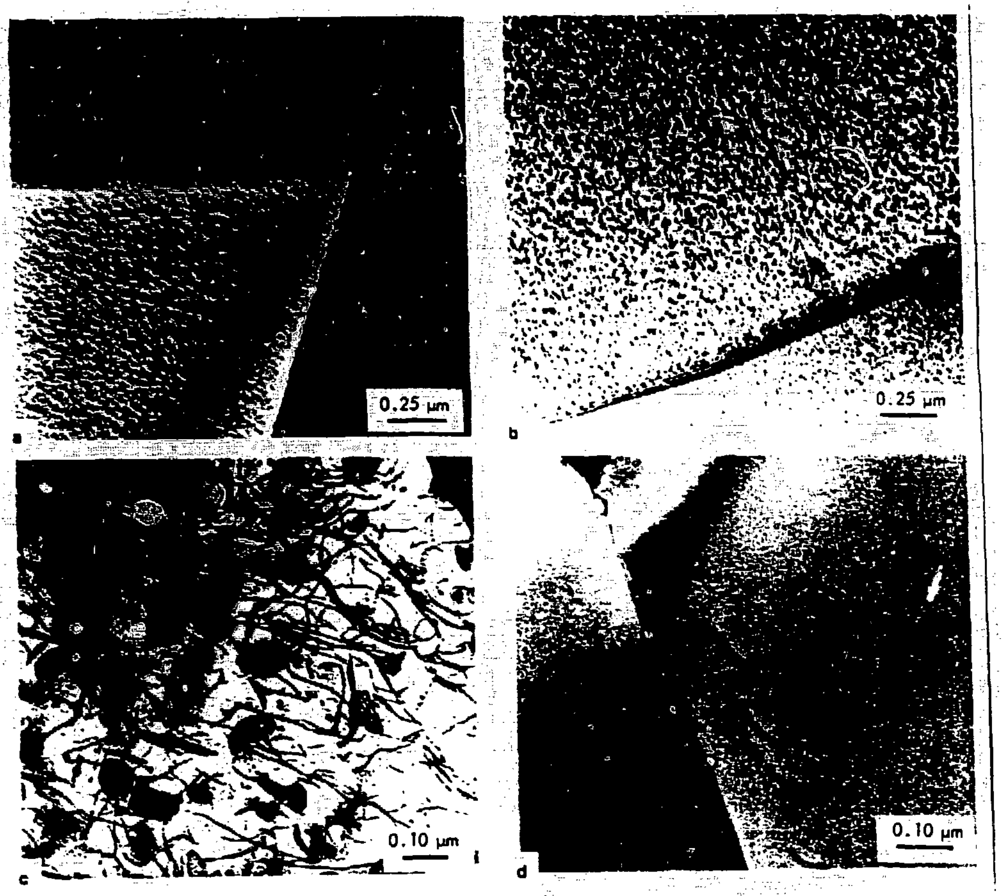

Fig. 2 — Defects produced in annealed 300-series stainless steel during neutron irradiation in a fast reactor, as observed by transmission electron microscopy. (a) Microstructure of unirradiated sample. (b) Interstitial dislocation loops formed during irradiation at 300°C (0.35 T_m). (c) Dislocation loops, voids, dislocation networks, and precipitates formed during irradiation at about 500°C (0.47 T_m). (d) Helium bubbles formed during irradiation at about 700°C (0.59 T_m).

Additional factors which influence the selection and development of structural materials relate to the fact that up to 20% of the energy produced in the plasma may be deposited on the first wall. The resultant heat flux creates a thermal stress which cycles in magnitude as the reactor power cycles. Under these conditions fatigue and fatigue crack growth can become life limiting properties for the structure. The magnitude of the thermal stress is related to the physical properties of the structural material through the so-called thermal stress factor $k(1 - \nu)/\alpha E$ where k is the thermal conductivity, α the coefficient of thermal expansion, ν Poisson's ratio and E Young's modulus.

As in all large power systems fabricability and weldability of the structural material will be important considerations. These characteristics may be even more important in a fusion reactor because of the complex geometry. Weldability will be important in the fabrication and in determining the feasibility of remote repairs to an activated assembly. Basic mechanical properties (along with compatibility and radiation damage) will be a primary contributor in determining the maximum operating temperature of the structural material. Neutron induced activation of materials in a fusion reactor affects three areas: reactor maintenance, engineered safety, and waste disposal. The issues central to development of low activation materials and the extent to which low activation materials might affect remote maintenance and waste management have been considered in datail.[2,3] Contact maintenance everywhere in the reactor system does not appear possible. This stems from the fact that in materials which are only weekly activated or in which the activation decays rapidly sufficient to permit contact maintenance the impurities could not be reduced to sufficiently low levels. It may be possible, however, to

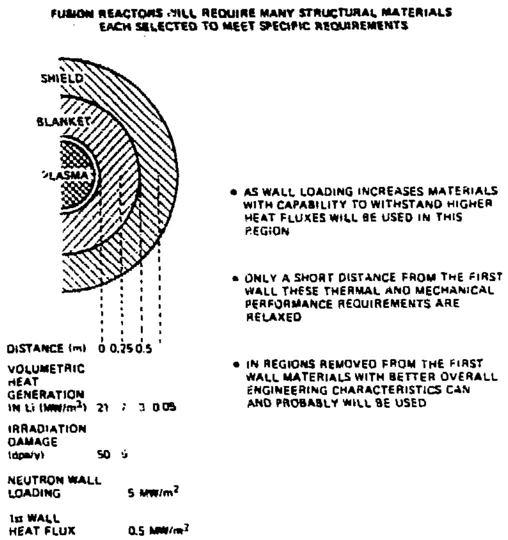

Fig. 3 — Approximate volumetric heat generation in lithium and displacement damage production in the structural alloy as a function of distance from the first wall.

significantly reduce the waste management burden. If certain elements are present in relatively small concentration deep geological storage may be required. Table 3 gives examples of approximate allowable initial concentrations of various elements in a material if it is to disposed of under the least restrictive rules for shallow land burial (i.e., Class C as defined by 10CFR61). Development of structural alloys which could be disposed of in shallow land burial rather than in deep geological storage has become a goal of the Fusion Materials Program. Austenitic stainless steels, ferritic steels, vanadium alloys, and ceramics such as SiC could be developed which, through restrictions on alloying elements and impurities, could be disposed of as low level waste by shallow land burial.

CANDIDATE STRUCTURAL MATERIALS

There have been approximately 20 different fusion power reactor concept studies.[4] Essentially all the possible combinations of tritium breeder, coolant, and structural material have been examined in varying degrees of depth of analysis. Numerous alloys or alloy systems have been suggested as a structural material. In this section we will briefly review the advantages and disadvantages of various alloy systems. The depth of our analysis will be greatest for advanced austenitic stainless steels, ferritic (martensitic) steels, and vanadium alloys. These three alloy systems currently represent the leading candidates.

Table 3 — Examples of Initial Concentration Level Restrictions* — 10CFR, Part 61 Waste Disposal Rules — Ten Years After Shutdown, 9 $MW \cdot y/m^2$ Exposure

Element	Limiting Radio-nuclide	Initial Concentration Limit,† Class C
N	^{14}C	3650
O	^{14}C	$>10^6$
Al	^{26}Al	$>10^6$
Co	^{60}Co	10^6
Cu	^{63}Ni	2400
Fe	^{55}Fe	$>10^6$
Ni	^{63}Ni	20,000
Mn	^{54}Mn	$>10^6$
Mo	^{94}Nb	3650
Nb	^{94}Nb	1
Zr	^{90}Sr	$>10^6$
Si	^{31}Si	$>10^5$
Mg	^{24}Na	$>10^6$
V	^{49}V	$>10^6$
Ti	^{45}Ca	$>10^6$
Cr	^{49}V	$>10^6$

*These limits apply to the first wall region.
†Atom parts per million based on assumed atom density of base materials of 8×10^{22} cm^{-3}.

ALUMINUM ALLOYS — Aluminum alloys have been proposed for use in low activation concepts in which the goal is to minimize induced radioactivity in the reactor structure.[5,6] A major limitation of aluminum alloys is their elevated temperature strength. The maximum operating temperature for wrought aluminum alloys is about 150°C. Powder metallurgy dispersion strengthened alloys would have acceptable properties to about 300°C. The difficulties of joining powder metallurgy alloys to produce a complex structure in which the welds have acceptable properties would have to be solved. Aluminum alloys are generally compatible with He and H_2O but incompatible with the liquid metals, viz., Li and 17 Li—83 Pb. To achieve the high coolant temperatures required for acceptable thermodynamic efficiency a design using two temperature zones in which the aluminum operated at acceptably low temperatures would have to be used. Thus a second structural material would have to be used in the "hot" regions of the reactor.

COPPER ALLOYS -- Copper alloys will have many applications in fusion power reactors. The combination of high electrical conductivity and strength at low temperatures leads to their selection as normal conducting magnets and as the stabilizer in superconducting magnets. Because of their excellent thermal conductivity they will most likely find application in components subjected to high heat fluxes such as limiters, diverters, neutral beam target plates, first wall armor, and beam dumps. They have been proposed for use as structural alloys in compact reactor designs in which the thermal and neutron wall loadings are high relative to more conventional designs. In these applications the general requirements are the same as in more conventional designs: chemical compatibility, radiation damage resistance, and adequate mechanical properties. Copper alloys are compatible with helium and H_2O but not compatible with lithium or 17 Li—83 Pb. Designs utilizing copper as the structural alloy would thus be limited to use of a solid breeder. Precipitation strengthened copper alloys could be used to temperatures of about 300°C. Thus like aluminum alloys, the thermal efficiency will be limited unless a design with two temperature zones is used. Oxide dispersion strengthened alloy could be used to significantly higher temperatures (>450°C). Like dispersion strengthened aluminum alloys these alloys present difficult joining problems. As concerns radiation damage, there is very little information in the regime of temperature and neutron fluence of interest for blanket applications. It is thus difficult to make lifetime projections. A major concern regarding the use of copper alloys is induced radioactivity. For disposal as Class C waste in shallow land burial an alloy would have to contain less than about 2400 appm Cu — clearly eliminating copper base alloys from consideration as reduced activation materials.

NICKEL BASE ALLOYS — Nickel base alloys have attractive high temperature strength properties. Based on tensile and creep-rupture properties in the unirradiated condition they could be used to temperatures of about 750°C. Their compatibility with Li and 17 Li—83 Pb is generally poor. With these coolants and breeders the temperatures would be limited to less than 450 and 400°C respectively. In this case the nickel base alloys would have no advantage over austenitic and ferritic steels. The most attractive application of nickel base alloys is in helium cooled systems where metal temperatures of at least 500—600°C would be required. Nickel base alloys appear to have two limitations. Severe radiation embrittlement, which manifests itself as very low ductility (in some cases zero) grain boundary fracture, occurs at temperatures above about 500°C. Waste disposal rules place low limits on the amount of nickel in an alloy — less than about 2 at. % to qualify for disposal by shallow land burial — thus nickel base alloys are inherently high activation.

AUSTENITIC AND FERRITIC STEELS AND VANADIUM ALLOYS

The alloy systems which have the most attractive combination of properties and appear to have the most potential for development as fusion reactor structural materials are advanced austenitic stainless steels, ferritic (martensitic) steels, and vanadium alloys. Table 4 lists the reference candidate alloys which have been in the research and development program for a number of years and related low activation versions of these reference alloys. Research and development on the low activation austenitic and ferritic steels was initiated in 1984 and is in the very early stages.

ADVANCED AUSTENITIC STAINLESS STEELS — A significant effort has been devoted to the development of austenitic stainless steels for liquid metal fast breeder reactor cladding and duct applications. As a result of this effort a significant data base exists for the properties of standard type 316 stainless steel as well as advanced alloys tailored for the breeder environment. The fusion program has used the breeder reactor experience as a starting point in the development of austenitic alloys tailored for the fusion environment. The improvement in performance of the fusion Prime Candidate Austenitic Alloy (PCA) over type 316 stainless steel is significant.

Of the three leading alloy systems the austenitic stainless steels are clearly the easiest to fabricate and to weld. For welding, no extraordinary care must be taken to prevent contamination of the weld and resultant degradation of properties, and no preheat or post-weld heat treatment is required. There are no significant concerns regarding the use of austenitic alloys in a hydrogen environment. Austenitic alloys are among the most resistant to hydrogen embrittlement, and hydrogen permeability is lowest of the three leading candidate alloy systems (Fig. 4). It should be possible to reduce permeation further by the formation of stable oxide films on the alloy.[7] The use of austenitic alloys in helium and water coolants would not be limited by corrosion/mass transfer; however, stress-corrosion in water is a concern. In lithium and 17 Li—83 Pb the maximum operating temperature at the coolant-metal interface would be limited to approximately 450 and 400°C, respectively, and in nitrate salts to about 500°C by compatibility considerations.

Table 4 — Leading Candidate Reference and Low Activation Structural Alloys

Alloy Class	Reference	Low activation
Austenitic stainless steels	Path A PCA (Fe-15Cr-15Ni-Mo-Ti)	Mn stabilized steels
Vanadium alloys	V-15Cr-5Ti V-20Ti VANSTAR 7	V-15Cr-5Ti V-20Ti
Ferritic steels	HT-9 MOD 9Cr-1Mo 2 1/4 Cr-1 Mo	Fe-Cr-W Fe-Cr-V

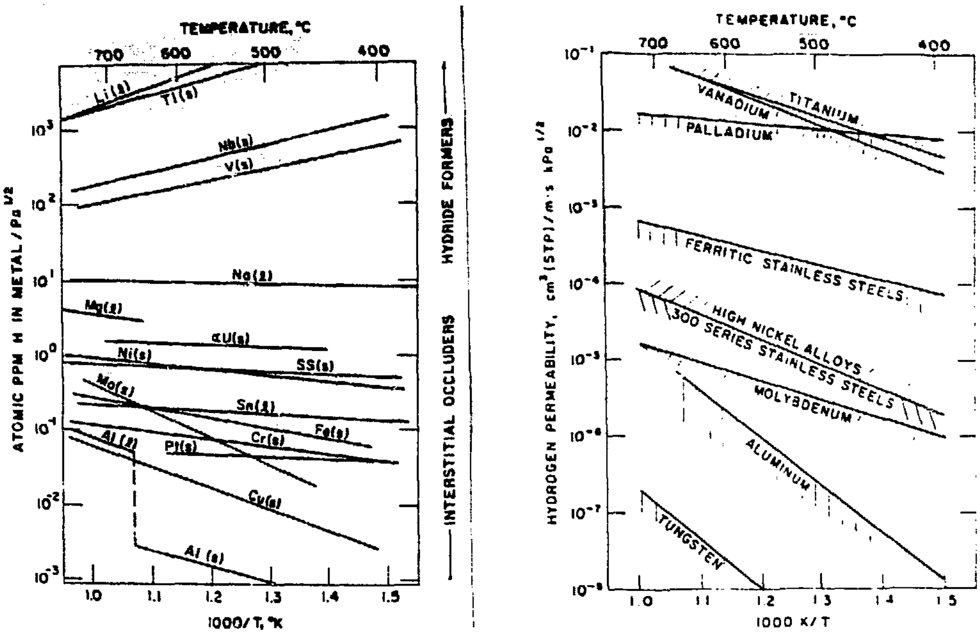

Fig. 4 - Hydrogen solubility and permeability in various alloy systems.

There are two primary limitations for the austenitic stainless steels. First a relatively low thermal stress factor ($K(1-\nu)/\alpha E = 4.8$), which translates to high thermal stresses and a limitation on the magnitude of a cyclic heat flux. The physical properties are inherent properties of the alloy system and thus the thermal stress factor cannot be changed substantially by alloy development.

The second limitation is radiation damage. Standard alloys such as type 316 stainless steel exhibit relatively large amounts of radiation induced swelling. A significant improvement in performance has been achieved for these alloys in the breeder reactor environment through relatively small modifications of the alloy chemistry.[8] This is illustrated in Fig. 5, which shows swelling as a function of damage level at an irradiation temperature of 500–650°C for 316 type stainless steel and an advanced stainless steel designed for breeder reactor applications. The swelling is reduced primarily by an extension of the incubation period. Similar alloy compositions are being developed for fusion applications.[9] The goal is to provide a high concentration of sites for nucleation of helium bubbles. If bubbles are nucleated on a very fine scale their growth is limited through competition for the available supply of helium atoms and vacancies. The time (or damage level) required to reach a critical size that would allow the bubbles to grow as voids thus leading to an acceleration in the swelling rate, is extended. Titanium carbide (TiC) precipitate particles are formed on a dislocation structure produced by cold working. These particles act as helium bubble nucleation sites. The result is shown in Fig. 6(a).

Loss of elevated temperature tensile ductility, creep ductility, and creep-rupture

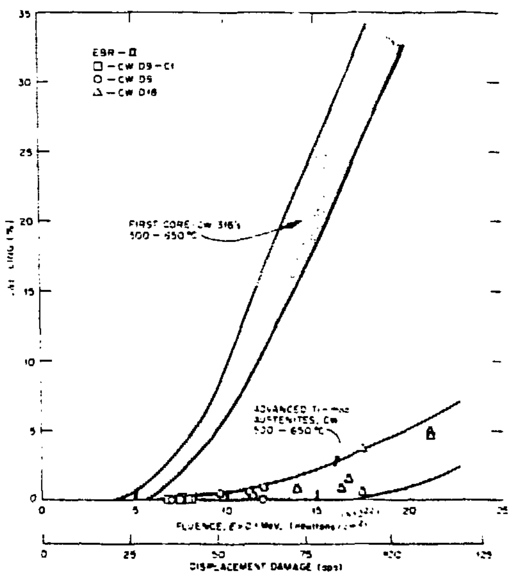

Fig. 5 - Swelling of type 316 and an advanced austenitic stainless steel which has been developed for breeder reactor applications.

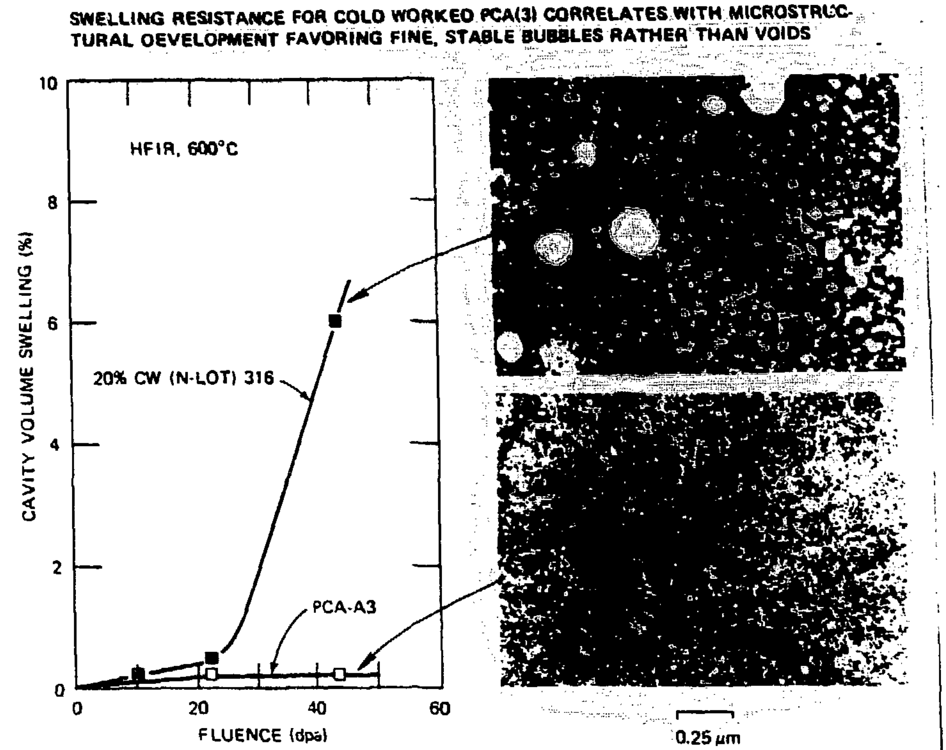

Fig. 6 - Swelling of type 316 stainless steel and an advanced austenitic stainless steel with composition and microstructure tailored for the fusion reactor environment. Irradiations were conducted in a mixed spectrum reactor and produced up to approximately 4000 appm He at the highest dpa level. This helium concentration is equalivant to approximately 27 MWy/m^2 exposure.

life also occurs as a result of neutron irradiation. These effects are at least in part attributable to the precipitation of helium bubbles at grain boundaries and the stress induced growth of these bubbles resulting in low ductility grain boundary fracture. As in the case of swelling, significant improvement is obtained through grain boundary precipitate structures which limit or retard the growth of bubbles at the boundary.[10] Figure 7 shows the ductility of an advanced austenitic stainless steel in three conditions: unirradiated, irradiated with a 20% cold worked microstructure, and irradiated with microstructure developed by aging to produce grain boundary TiC particles and then cold working 20%. In the 20% cold worked condition the ductility at 600°C was reduced to about 1% at a damage level of 22 dpa and 2000 appm He (a helium level equal to 10 MWy/m^2 at the first wall). For the same damage level the ductility remained at an acceptable 5–8% at 600°C for the grain boundary precipitate, 20% cold worked structure. The performance of the austenitic alloys relative to radiation damage has been improved significantly through alloy development. Lifetimes of 10–15 MWy/m^2 at temperatures approaching 600°C should be achievable from the standpoint of radiation effects.

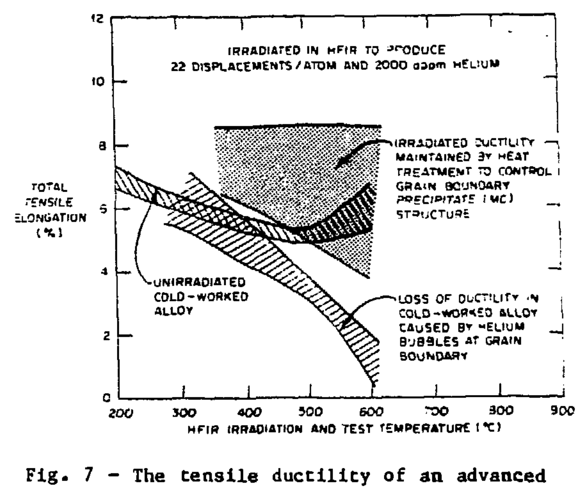

Fig. 7 - The tensile ductility of an advanced austenitic stainless steel is significantly improved by control of the grain boundary structure. Irradiations were performed in a mixed spectrum reactor and produced about 2000 appm He equivalent to 15 MWy/m^2 exposure in a fusion neutron spectrum.

FERRITIC (MARTENSITIC) STEELS

Ferritic steels offer the advantage of lower thermal stresses and slightly better compatibility with Li and 17 Li– 83 Pb relative to austenitic stainless steels. On the basis of results obtained from irradiations in fast reactors, shown in Fig. 8, the ferritic steels exhibit lower radiation induced swelling.[11] The incubation period for the onset of swelling is beyond 100 dpa which is much longer than that of type 316 stainless steel. The swelling rates of the binary alloys are lower than those of simple austenitic alloys suggesting that the ferritics may swell at an inherently lower rate once incubation occurs. Recent results from mixed spectrum reactor[12] and ion irradiations suggest[13] that the increased helium generation rates in the fusion spectrum will reduce the incubation period. The swelling rates may, however, remain lower than in austenitic alloys.

The two dominant concerns regarding the use of ferritic steels relate to observed increases in the DBTT above room temperature and to weld procedure requirements, viz., postweld heat treatment (PWHT). Increases in the DBTT to ~125°C have been observed after irradiation at 400°C.[14] Although not confirmed for HT-9, increases in the DBTT to ~280°C have been observed for pressure vessel steels when irradiated at 250—300°C. These temperatures are within or near the anticipated operating temperature range for all coolants. The effects of relatively high helium generation rates in a fusion spectrum may further exacerbate this problem. Although acceptable weld procedures have been developed for HT-9, the reliability of large numbers of complex welds anticipated in most blanket designs remains a concern. The potential effects of using a ferromagnetic material in a magnetically confined fusion reactor has not been fully evaluated. Preliminary analyses indicate the magnetic loadings may be significant but not excessive.[15] Thermal creep will impose constraints on HT-9 at temperatures above 500°C. This may be a more restrictive lifetime constraint than swelling in high stress regions created by high pressure coolants.

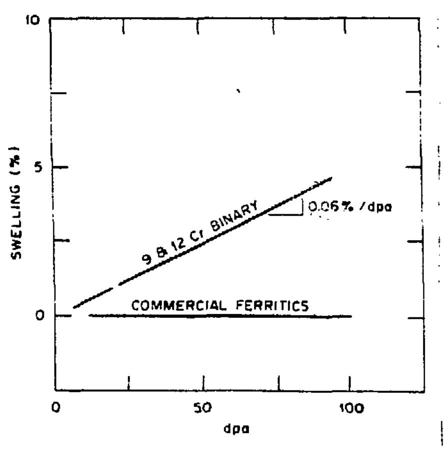

Fig. 8 - Swelling of simple binary Fe-Cr alloys and commercial ferritic steels as a result of irradiation at 450°C in a fast reactor spectrum.

VANADIUM ALLOYS

Vanadium alloys have four important attractive features. (1) They have the lowest thermally induced stresses of the three leading candidate alloy systems. (2) On the basis of their elevated temperature strength properties they could be used at higher temperatures than either austenitic or ferritis steels. (3) The base alloy systems (e.g., V-Cr-Ti, V-Ti-Si) are the most attractive from the viewpoint of neutron activation. (4) They have good compatibility with liquid metals. On the basis of results from fast reactor irradiation experiments they appear to have better radiation damage resistance than austenitic stainless steels both from the viewpoint of swelling and embrittlement. However, as with the ferritic steels very little helium is produced in the fast reactor spectrum. Recent results from neutron irradiation experiments in which helium was preinjected with an accelerator to levels more appropriate to fusion service show that the V– 20 Ti alloy exhibits radiation induced embrittlement.[16] Swelling at fusion reactor damage levels has been investigated with dual beam ion irradiations. At 125 dpa and 9.3 appm He/dpa (equivalent to about 11 MWy/m^2) negligible swelling was found indicating that vanadium alloys may retain their low swelling characteristics in the fusion environment.[17]

The major concerns regarding the use of vanadium alloys are oxidation and interstitial contamination and the solubility and permeability of tritium in the alloys. Vanadium alloys are severely embrittled when contaminated with interstitials such as oxygen and nitrogen. Fabrication processes, including welding, must be performed in an inert environment to avoid contamination. Unacceptable oxidation of the alloy would occur if exposed to even low partial pressures of oxygen at elevated temperatures.

Tritium permeation would be a major concern in H_2O cooled systems. Vanadium is not considered compatible with helium coolant because of impurities (H_2O, O_2) in the helium. Vanadium alloys offer the greatest potential for lithium cooled systems because of their excellent compatibility with liquid lithium if interstitial impurity effects can be controlled. If the primary coolant only contacted vanadium alloys, operating temperatures as high as 750°C might be achieved (assuming radiation damage does not limit). However, if the primary circuit were a mixed system (e.g., a stabilized ferritic steel intermediate heat exchanger and a vanadium alloy blanket system) the maximum temperatures may be considerably lower, because of interstitial mass transfer.

The V-15Cr-5Ti alloy exhibits most of the attractive properties associated with the vanadium alloy systems and is currently considered the reference alloy. Further research is required to optimize the composition and thermal mechanical treatment for the fusion reactor application. Limited fatigue and creep data for the vanadium alloys indicate superior performance compared to the steels.[18]

SUMMARY

The blanket is one of the largest and most complex systems in a fusion power reactor. The primary functions of the blanket system are recovery of heat produced in the plasma and breeding of tritium for the reactor fuel cycle. Performance of the blanket in terms of thermodynamic efficiency of the power conversion cycle, reliability, tritium breeding and lifetime of the system will be major factors in determining the cost of electricity from a fusion power reactor.

Numerous possibilities exist from which a compatible combination of blanket structural material, coolant, and breeding media must be selected. It is not possible at this time to establish which combination is optimum. Significantly more information will be required on the performance of materials, particularly the structural material and breeding media, before this can be accomplished. If one selects a structural material then the choice of breeding media and coolant is constrained and limits can be placed on the maximum temperature of the structural material.

The primary constraints and limits for six classes of structural alloys are summarized, as we presently know them, in Table 5. For example, if one selected a vanadium alloy as structural material and water (steam) as coolant, then the maximum metal temperature would be limited to ~400°C by structural alloy — coolant compatibility. However, if the system were all vanadium with lithium coolant the maximum metal temperature would be >650°C from consideration of radiation effects.

The primary favored candidate structural materials are the austenitic stainless steels, ferritic (martensitic) steels, and vanadium alloys. In the past few years significant progress has been made towards understanding the factors which will limit performance of these alloy systems. A major factor is radiation damage. The key to improving radiation damage resistance appears to be preventing transmutation produced helium from agglomeration into helium bubbles of sufficient size to undergo unstrained growth as voids (i.e., rapid radiation induced swelling) and to prevent the helium from forming large bubbles at grain boundaries in which case it is a major contribution to grain boundary embrittlement. In the case of austenitic stainless steels large improvements in radiation damage resistance have been obtained through control of microstructure and composition — in particular the trapping of helium on a very fine scale at TiC precipitates and on dislocations. The principles developed for improving performance of the austenitic stainless steels can be applied to other alloy systems. Alloys with improved properties for fusion power reactor applications can be developed.

Table 5 — Temperature Limits (°C) for the Structure Material Determined from Consideration of Compatibility, Mechanical Properties, and Radiation Response

Alloy	Maximum Metal Temperature (°C) as Determined by						
	Compatibility with Coolants				Mechanical Properties	Radiation Effects[a]	
	H_2O	He	Li	Li-Pb		Temp (°C)	MWy/m^2
Austenitic stainless steel							
316	NL	NL	450	400	650	550	5
Advanced fusion alloy	NL	NL	450	400	650	600	10–15
Ferritic steels							
modified 9Cr-1Mo or HT-9	NL	NL	500	425	550	550[c]	>10
Vanadium alloys	400	400	750[b] 550	650	750	650	?
Copper alloys	300	NL	NC	NC	300	200?	?
Aluminum alloys (wrought)	NL	NL	NC	NC	200	150?	?

NC = not compatible
NL = Clearly not limiting
[a] Where possible the lifetime at the first wall is estimated. Lifetime is at the radiation effects temperature limit and is based on either swelling or embrittlement.
[b] Compatibility temperature limit for an all V, Li cooled system. If mixed system (V Blanket — Ferritic Steel IHX) the limit would be 550°C.
[c] Shift in DBTT may place lower limit on operating temperature which is in the range 250–300°C.

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