MASTER

COMPATIBILITY OF STRUCTURAL MATERIALS WITH FUSION REACTOR COOLANT AND BREEDER FLUIDS*

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Fusion reactors are characterized by a lithium-containing blanket, a heat transfer medium that is integral with the blanket and first wall, and a heat engine that couples to the heat transfer medium. A variety of lithium-containing substances have been identified as potential blanket materials, including molten lithium metal, molten LiF-BeF_2 , Pb-Li alloys, and solid ceramic compounds such as Li_2O . Potential heat transfer media include liquid lithium, liquid sodium, molten nitrates, water, and helium. Each of these coolants and blankets requires a particular set of chemical and mechanical properties with respect to the associated reactor and heat engine structural materials. This paper discusses the materials factors that underlie the selection of workable combinations of blankets and coolants. It also addresses the materials compatibility problems generic to those blanket-coolant combinations currently being considered in reactor design studies.

1. INTRODUCTION

The production of tritium and the conversion of neutron kinetic energy to sensible heat are integral to the functioning of a fusion energy system. Accordingly, fusion reactors are characterized by a blanket region used to thermalize neutrons and produce tritium, a heat transfer medium that is integral with the blanket and first wall, and a heat engine that couples to the heat transfer medium.

One of the principal areas of dialogue between fusion reactor designers and the materials community has concerned the compatibility of first-wall materials with potential blanket and heat transfer concepts. This paper reviews the current status of this dialogue, namely, the impact on reactor design stemming from chemical interactions between the first wall and candidate blanket-coolant systems.

1.1 Breeding Materials for Fusion Reactors

In current D-T fusion reactor concepts, tritium is bred by neutron absorption in a lithium-containing blanket immediately surrounding the plasma. Whatever the breeding material, the blanket must satisfy at least four primary requirements: (1) neutron absorption by lithium nuclides must occur to the exclusion of competing reactions [except for $(n,2n)$ reactions]; (2) the breeding material must resist thermal or radiation-induced decomposition; (3) tritium must be recoverable from the blanket by a modest investment and to a level that results in a low tritium inventory in the blanket; and (4) those alloy

families that can be used to contain the breeding material must also be acceptable as first-wall materials.

The blanket serves as a heat source as well as a tritium generator and must be designed to accommodate the transfer of high-grade thermal energy to the power conversion system. In the case of magnetically confined fusion reactors (MCFRs), the use of electrically conducting liquids as blanket materials is complicated by magnetohydrodynamic (MHD) effects, which are produced by the movement of the liquids across magnetic field lines [1]. Accordingly, blanket concepts for MCFRs have envisaged the use of a solid or semi-stagnant liquid blanket cooled by a nonconducting liquid or gas. In contrast, inertially confined fusion reactor concepts characteristically integrate the breeding and cooling functions into one "homogeneous" lithium-containing circuit.

Of the liquid blanket materials proposed to date, candidates have narrowed to lithium, Pb_4Li , and Li_2BeF_4 . Solids have included Li_2O , LiAlO_2 , LiAl , and Li_7Pb_2 .

1.2 Coolant Materials

Some of the energy conversion systems proposed [2] in conjunction with MCFR blankets are listed in Fig. 1. These various concepts embody a relatively wide range of coolant forms and compositions:

Coolant Material	Physical Form
Na, Li, Pb-Li eutectic	Liquid metal
$\text{NaNO}_3\text{-KNO}_3\text{-NaNO}_2$	Molten salt
He, CO_2 , steam	Gas
H_2O , K	Liquid and gas

As discussed by Fraas [3], the designer must satisfy several constraints in selecting a fusion reactor coolant. However, this review

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Fig. 1. Generic combinations of blanket, coolant, and power cycle proposed for fusion reactors.

will focus on only one of the constraints — namely, the chemical compatibility of the coolant with the first wall and blanket. This assessment of the compatibility of prospective blanket-coolant systems will also be based on a maximum temperature requirement of approximately 600°C. The latter temperature matches the currently estimated first-wall temperature for a first-generation fusion reactor, and is limited by radiation damage (i.e., helium embrittlement) [4].

1.3 Issues in Assessing Compatibility

The compatibility of first-wall materials with blanket or coolant materials has been judged almost exclusively in terms of the "wastage" or thinning of the first wall. On this basis, a corrosion rate of approximately 25 $\mu\text{m}/\text{year}$ (~ 1 mpy) is usually taken as the upper bound for acceptable blanket or coolant compatibility. However, other corrosion effects may militate for an even lower corrosion limit. Chief among these effects are the selective leaching of certain elements contained in the first wall, interstitial impurity transfer to or from the wall, and the accumulation of corrosion products in heat exchanger tubes. Another important corrosion consideration is the degree of mass transport of radioactive first-wall elements to piping and heat exchanger surfaces outside the reactor radiation shield. This transport is particularly significant if it requires remote maintenance for regions outside the nuclear island.

2. BREEDER-BLANKET STRUCTURAL METAL COMPATIBILITY

2.1 Liquid Lithium

Among the prospective breeding materials for fusion reactors, liquid lithium serves as the reference against which alternatives are judged. The MHD problem associated with the use of lithium in magnetically confined machines is its major drawback [5]. Yet serious questions also remain regarding its compatibility with first-wall materials, principally because the available corrosion data for lithium with most alloys are limited. (Substantial corrosion information does exist for niobium- and tantalum-base alloys in liquid lithium.)

Table 1 sets the lithium compatibility question in context. Several candidate first-wall materials are listed, together with estimates of the highest temperatures at which these materials could be used in contact with lithium. The upper value is an estimate of highest operating temperature that we are likely to achieve given the 25 $\mu\text{m}/\text{year}$ wastage rate as the limiting criterion. The lower value is a more pessimistic estimate, which disallows any reduction in currently measured corrosion rates and sets up criteria other than wastage rates as more constraining. The latter criteria include the deposition of corrosion products, interstitial impurity mass transport, preferential grain boundary attack, and radioactive isotope mass transport. It must be emphasized that the temperatures listed (except for niobium alloys) represent very rough estimates and, in the case of nickel-base alloys, the pessimistic value cannot even be estimated with present data.

Table 1

Estimated temperature constraints imposed by liquid lithium compatibility with prospective fusion reactor structural materials.

Candidate Structural Material	Temperature Limit Estimate, °C	
	Lower	Upper
Stainless steel	500	600
Nickel-base alloy	?	500
Vanadium alloy ^a	650	?
Titanium alloy	700	?
Niobium alloy ^a	650	1000

^aIn combination with austenitic stainless steel.

2.1.1 Lithium Compatibility with Structural Steels

The corrosion of iron-base alloys by lithium has been studied at Oak Ridge National Laboratory in slow-flowing, naturally convecting loop systems [6]. Although the rates depend on the loop geometry, the results indicate that, even at the relatively low flow rates in these systems, the imposition of a temperature gradient will cause steady-state dissolution rates of up to 12 $\mu\text{m}/\text{year}$ (0.5 mil/year) at a maximum temperature of 600°C.

In the case of austenitic stainless steels, dissolution by the lithium results in a depletion of chromium and nickel and enrichment of iron at the hotter loop surfaces [6]. Steady-state surface concentrations ultimately develop as the iron, chromium, and nickel dissolution rates become proportional to the bulk concentrations of these elements in the stainless steel [7]. At 500 to 700°C, austenitic stainless steels containing 18% Cr and 8 to 10% Ni undergo a transformation to ferrite near the exposed surface; this transformation zone reaches a steady-state thickness from 20 to 60 μm within 3000 h at 600°C [6]. Under the latter conditions, the corrosion rate appears to be largely controlled by the mass transfer rate of iron. Thus, the corrosion rate of the austenitic stainless steels at steady state approaches the rate for ferritic steels such as 2 1/4 Cr-1 Mo. However in the case of 18% Cr-10% Ni stainless steels, the chromium poses a significant problem in terms of cold leg deposition. The chromium deposits as needle-like metallic crystals, which because of their high specific surface area impede the lithium flow, even when the deposit density is relatively low. Chromium deposition is expected to be less troublesome in the case of ferritic steels with 2.2 to 9% Cr.

Corrosion tests with lithium at higher flow rates than the tests described above are being conducted in a type 316 stainless steel loop at Westinghouse Advanced Reactor Division (WARD) [8]. Corrosion rates at the 538°C maximum temperature in the WARD loop averaged 20 $\mu\text{m}/\text{year}$ between 200 and 948 h. This rate is significantly higher than the 12 $\mu\text{m}/\text{year}$ for thermal convection loops at 600°C and suggests that a lithium blanket-coolant system under forced flow will corrode substantially faster than a semi-stagnant lithium blanket cooled by an independent heat transfer circuit.

However, to the extent that hydrodynamic factors govern the mass transport of steel components in lithium, the strong magnetic fields in MCFRs may also bear on the corrosion rate. The corrosion properties of lithium in stainless steel systems are qualitatively very similar to those of sodium in stainless steel. In the latter system the thickness of the boundary layer near the loop wall where the sodium flow is laminar appears to control the

mass transport rate of iron down to a boundary layer thickness of approximately 0.5 mm [9]. (At lesser thicknesses the rate becomes reaction-rate limited.) In considering the effect of transverse magnetic fields on lithium flow, Wells [10] has noted that, as the magnetic field acts to restrict turbulence, the effective boundary layer or Hartmann layer is much thinner than the laminar boundary layer produced under more turbulent flow. Thus, if diffusion in the liquid lithium controls the dissolution rate of iron, the corrosion rate would be expected to increase at a given mass flow with the imposition of a transverse magnetic field. So far, however, no experimental measurements have been made to verify this effect in lithium (or any other liquid metal).

The knowledge gained from corrosion studies of lithium is now being applied to corrosion inhibition techniques. Nitrogen in lithium increases the weight loss and intergranular attack of types 304L [11] and 316 stainless steel [12] under stagnant conditions. (However, the nitrogen effect appears much less significant in flowing, nonisothermal systems [13].) The addition of small amounts of calcium to lithium effectively neutralized nitrogen-induced corrosion of type 316 stainless steel at 500°C [12]. Aluminum also shows promise as an inhibitor addition to lithium. In addition to being a nitrogen scavenger, dissolved aluminum reacts with the nickel in stainless steel to form a thin, highly protective intermetallic coating at the interface between lithium and stainless steel. A piece of aluminum metal weighing 5% of the lithium inventory was included with the hot-leg specimen string in a type 316 stainless steel lithium thermal-convection experiment operated with maximum and minimum temperatures of 600 and 400°C. The weight loss of stainless steel coupons at 600°C was decreased by a factor of 5 compared with a control test with no aluminum [14]. Although the aluminum addition appears extremely promising as a technique for corrosion inhibition, questions remain as to the effect of aluminum on the behavior of tritium in lithium.

2.1.2 Lithium Compatibility with Other Structural Materials

As already noted, corrosion tests with lithium have been confined principally to iron-, niobium-, and tantalum-base containment materials. Early screening tests indicated that nickel-base alloys were not compatible with static lithium at 800°C; accordingly Fe-Cr-Ni alloys containing more than 15% Ni have generally not been considered for lithium containment. However, this has resulted in an information gap as to the lowest acceptable temperature level at which higher nickel Fe-Ni-Cr alloys could be used in contact with lithium (Table 1).

Corrosion data for titanium, zirconium, and vanadium are also quite limited, but each of

these metals appear relatively resistant to corrosion and their all-gradient mass transfer by lithium, at least up to the temperature limits set by mechanical property considerations. A major concern with these relatively reactive metals is their tendency to "getter" interstitial impurities such as nitrogen, carbon, and hydrogen from the lithium, an effect that can seriously embrittle the metal. Carburization and nitriding are of particular concern in the case of vanadium alloys, since these alloys would have to be used in combination with a more oxidation-resistant alloy to interface with the steam-driven power conversion system. Steam-exposed components constructed of austenitic or ferritic stainless steels will require a source for carbon and possibly hydrogen when used in a common lithium circuit with vanadium alloys. Although extremely corrosion resistant to lithium, niobium alloys also are subject to carburization and nitriding if used in a common circuit with conventional iron-base alloys. The use of a stabilized grade, such as type 321 stainless steel, is known to reduce the extent of carbon transport to Nb-Fe-Zr in sodium and potassium circuits up to about 600°C [15], but the effect of stabilization has not yet been investigated in dissimilar-metal lithium circuits.

2.2 Molten Salts (LiF-BeF_2)

Among the lithium-containing molten salts of potential interest as breeding and cooling fluids, only the fluorides offer the high chemical binding energies necessary for radiation stability along with reasonable tritium breeding ratios and the potential for tritium recovery. Extensive testing in support of fission reactor programs has provided a detailed data base on the chemical and physical properties of molten LiF-BeF_2 . The chemical stability of this salt makes it particularly attractive for blanket applications, where chemical compatibility with structural and moderator materials is essential. Also, the limited solubility of tritium in the salt greatly facilitates tritium removal. The primary disadvantages of the salt are (1) its relatively high melting point (364°C) and (2) its breeding properties, which may require the use of a neutron multiplier in concert with the salt blanket.

The corrosion properties of LiF-BeF_2 mixtures have been studied over a wide range of temperature and flow conditions. The mixtures are relatively unreactive toward such metals as Ni, Mo, and Fe. However, more reactive alloying elements such as Cr, Al, Ti, Nb, and V are oxidized by LiF-BeF_2 melts unless highly reducing conditions can be maintained in the melt. Thus, alloys such as the nickel-base Hastelloy N and the molybdenum based TZM resist corrosion in these salt mixtures up to temperatures at which the strength of the alloy becomes limiting (700°C or above). Alloys

containing more than 10% Cr will require some form of chemical control to limit the oxidation potential of the fluoride mixture. Acceptable corrosion rates of type 316 stainless steel at 600°C have been attained in $\text{LiF-Tl}_2\text{O}_3$ systems where the redox potential of the salt was buffered by a beryllium reductant [16]. The redox potentials required to accommodate alloys based on niobium and vanadium appear impractical, given the unique operating features of fusion reactors discussed below.

There are important unknowns in assessing the corrosion behavior of fluoride salts relative to fusion reactor blanket applications. The fraction of tritium carried as TF compared with that carried as T_2 will establish the redox potential of the salt; thus the TF/T_2 ratio of the salt must be tailored to the chemical reactivity of the container metal. Transmutations of beryllium and fluorine in the blanket will also yield oxidants capable of corroding structural metals. Thus, techniques for buffering the redox potential of the salt would almost certainly be required both to maintain a desired TF/T_2 ratio and to control the corrosion effects of transmutation reaction products.

In electrolytes flowing across strong magnetic fields, the Lorentz forces acting on the ionic components cause charge separation. The effect is to induce an emf in the system, which affects both the free energy and kinetics of chemical reactions occurring between the salt and its container wall. The magnitude of the emf depends on the ionic character of the salts, dimensions of piping cross sections, and flow velocities. Despite the expected importance of magnetic fields on the chemical compatibility of a molten salt blanket, the effect cannot be quantitatively assessed without more definitive reactor designs and electrochemical data on salt-metal systems.

2.3 Solid Components

Of the solid lithium-containing compounds suggested as breeding materials, only Li_2O has been studied experimentally with respect to chemical compatibility. In compatibility tests of Li_2O with Fe-Ni-Cr alloys between 800 and 1100°C, Takeshita et al. [17] report finding reaction products of the type LiCrO_4 and Li_5FeO_4 . It is significant that the latter compound was produced even at relatively low oxygen activities above the Li_2O , namely, under high vacuum. However, data have not yet been reported as to the effect of the observed reactions on the structural integrity of the container material.

3. COOLANT COMPATIBILITY

3.1 Liquid Metals

Molten lithium and sodium have each been considered as coolants for fusion reactors.

The use of lithium as a coolant as well as blanket eliminates the need for an intermediate heat exchanger between the blanket and coolant circuits. Sodium is of interest because its properties have been thoroughly studied in association with the fast breeder (fusion) reactors, and it can be contained in conventional Fe-Ni-Cr alloys at somewhat higher temperatures than those projected for lithium. The same compatibility criteria as used in estimating containment temperatures for prospective breeding materials lead to temperature limits of 700 and 600°C in austenitic stainless steel sodium and lithium circuits, respectively. If the lithium is semi-stagnant, as would be the case for a lithium blanket cooled by sodium, a somewhat higher lithium temperature could be tolerated than for a higher velocity lithium blanket-coolant circuit (Sect. 2.1.1). However, the temperature drop required to transfer heat from the lithium to sodium would probably usurp any potential gain in coolant temperature over that of the all-lithium system.

3.2 Molten Salts

Molten salts as a class afford relatively high boiling points and, hence, can be contained at relatively low static pressures. Their electrical resistivities are higher than those of liquid metal coolants, so that they can be moved across strong magnetic fields without incurring the strong braking action induced in liquid metals. However, those molten salt systems that are known to provide acceptable chemical and radiation stability have characteristically high melting points ($\geq 300^\circ\text{C}$). Also, the electrical potential induced by the movement of molten salts through magnetic fields poses an important unknown in assessing their chemical stability (Sect. 2.2). Table 2 lists molten-salt systems that have been under consideration as coolants for fusion reactors.

Table 2
Potential molten-salt coolants.

Type of Salt	Melting Temperature ($^\circ\text{C}$)	Advantages	Disadvantages
$\text{FNO}_3\text{-NaNO}_3\text{-KNO}_2$	140	Low melting point, inexpensive, compatibility with stainless steel	Radiolytic decomposition, reaction with Li
NaOH-KOH-LiOH	160	Low melting point, inexpensive, radiation stability	Corrosive, reaction with Li
Chlorides ($\text{ZnCl}_2, \text{AlCl}_3$)	100-450	Low melting point, radiation stability, limited reaction with Li, possibility of nonionic bonding	Corrosive

Nitrate-base fused salt mixtures have been widely used as heat transport fluids and for metallurgical heat treating. Although detailed

corrosion data are not available, AlCl_3 salt ($0.5\text{M NaNO}_3, 1\text{M NaNO}_2$, and 13M FNO_3) has been suitably contained within its useful temperature range ($450-500^\circ\text{C}$) by stainless steel and iron-base alloys with corrosion rates not exceeding 20 mg/year (1 mpy) [16]. The behavior of molten nitrate-nitrite salts under irradiation has not been experimentally determined; however, thermal decomposition of salts becomes significant above 500°C , and it is likely that radiation would promote decomposition at still lower temperatures. When lithium and HNS are mixed, vigorous reactions are expected in which the nitrate-nitrites would be reduced to LiH and Li_2N , the latter being extremely corrosive [16]. The reaction rate and the exothermic energy released would be a safety limitation.

The most serious limitation of the hydroxides as coolants is their corrosivity. No satisfactory material has been found for containing hydroxides above about 600°C . Below 600°C , alloys based on nickel and molybdenum appear to withstand molten NaOH , but only a few experiments have been run to verify this. Corrosion in hydroxide melts occurs by direct oxidation of the container metal to an oxy-salt accompanied by the reduction of hydroxyl ions to form hydrogen and oxide ions [19]. Complicating side reactions can include the formation of a metal hydride, the reduction of alkali metal ions, and the production of water. The corrosion of nickel by fused sodium hydroxide can be inhibited at 800°C by a 0.1 MPa (1 atm) overpressure of hydrogen [20]. Leakage of steam into hydroxides is tolerable from a safety standpoint but would produce an environment conducive to stress-corrosion cracking of iron- and nickel-base alloys.

Molten chlorides are generally similar to fluorides in their corrosive attack on metals. Thus, the discussion in 2.2 is directly relevant. The lower-melting chloride systems based on ZnCl_2 or AlCl_3 have relatively low binding energies and, hence, should be relatively corrosive. Acceptable container materials would probably be limited to graphite, nickel, and molybdenum.

3.3 Helium

The physical and chemical properties of pressurized helium make it an attractive coolant for magnetically confined fusion reactors. Helium is compatible with many structural alloys and blanket materials, is not subject to NHD effects, does not present a significant radiation hazard, and affords favorable chemical characteristics for tritium containment and processing. In addition, a well-developed technology base exists for helium from its use as a coolant for fission reactors. Although helium itself is chemically inert, impurity gases carried by the helium tend to be chemically reactive with blanket and power conversion loop components. The impurity

content of the helium will be determined by in-leakage, desorption from loop components, diffusion through the helium containment from the plasma and power conversion systems, transmutation, chemical reaction of impurities with each other or other materials in the loop, and the efficiency of the coolant purification system. The more probable impurities include hydrogen isotopes, H_2 , D_2 , CO , CO_2 , CH_4 , and N_2 .

The limitations on service temperatures posed by impurities in helium are indicated for various classes of structural materials in Table 3. Considerable data have been accumulated during the last 20 years concerning the interaction of helium impurities with austenitic stainless steels [21] and nickel-base superalloys [21, 22]. As indicated in Table 3, austenitic stainless steels effectively resist carburization and oxidation by normal impurities in helium up to about 750°C. Certain nickel-base superalloys exhibit significant carburization and internal oxidation above 800°C in helium containing principally H_2 , H_2O , and CO impurities, although alloying elements such as Ti and Si in conjunction with Cr appear to reduce the rate of corrosion. In any case, the acceptable temperature ranges for these materials are well above the 600°C limit currently targeted for first-generation fusion reactors.

Table 3
Estimated temperature constraints imposed by coolant compatibility with structural materials for a helium-cooled solid lithium compound blanket.²

Candidate Structural Material	Temperature Limit Estimate, °C	
	Lower	Upper
Stainless steel	750	?
Ni-Cr-Fe alloy	800	?
Titanium alloy	?	550
Vanadium alloy	350	450
Niobium alloy	400	500

²The blanket material is presumed to be canned in the structural material.

In contrast, the operating temperatures for reactive and refractory metals in "reactor-grade" helium are much more restricted by compatibility considerations. There are no known techniques for purifying helium to the extent required to prevent contamination of vanadium, niobium, and presumably titanium alloys during prolonged exposures at 600°C. To assess the interstitial contamination problem requires an estimate of the mass and

surface area of the helium containment system as well as the leakage rates of impurities into the system (circulators and steam generators being the principal sources). The estimated temperature limits shown in Table 3 for vanadium, niobium, and titanium alloys are based on the performance of these materials in vacuum with residual impurities at a level of 1 mPa (10^{-2} torr). In a typical helium coolant circuit, such impurity partial pressures translate to volume concentrations that are well below one part in 10^9 and represent an extreme target in terms of in-leakage restrictions and gas purification. Partial pressures of hydrogen and tritium in helium are also of concern, especially in the case of titanium alloys. Current aerospace specifications limit hydrogen concentrations in Ti-6Al-4V to 150 wt ppm, [23] which at 500°C corresponds to an equilibrium hydrogen (tritium) pressure of approximately 1.3 Pa (0.01 torr). However, still lower tritium pressures would be required to meet the 150 ppm H (T) specification in the colder regions of a titanium alloy helium circuit, and these lower pressures could ultimately fall below those needed for tritium recovery.

3.4 Water

The experience and data base gained in the use of water as a coolant and working fluid far exceed those for any other material. The compatibility of structural materials such as austenitic stainless steels, nickel-base alloys, and titanium is well documented for pressurized water or water-steam mixtures. Virtually no information exists, however, on the compatibility of niobium or vanadium alloys with high-temperature ($\geq 300^\circ\text{C}$) water.

The most serious corrosion problems in water circuits are associated with evaporation, since the expulsion of steam during boiling can produce high concentrations of nonvolatile corrosive impurities in the water phase. Thus the control of water chemistry, principally through the removal of undesirable impurities and the addition of chemical inhibitors, is fundamental to the operation of water circuits, irrespective of the containment material. In the presence of high-level radiation fields, the dissociation of water into hydrogen and oxygen further complicates the water chemistry problem. Because there are no presently accepted techniques for effecting oxygen recombination in boiling water circuits, fission reactors cooled by conversion of water to steam (BWRs) typically operate with dissolved oxygen concentrations as high as 0.2 wt ppm. Such oxygen concentrations have produced stress-corrosion cracking of weld-sensitized austenitic stainless steels under cyclic or steady-state loading. In the case of fission reactors cooled by pressurized water (PWRs), a hydrogen overpressure is maintained to effect oxygen recombination. The efficiency

of this approach in the case of fusion reactors will depend on the degree to which hydrogen from the water phase would tend to degrade the D-T plasma reaction.

4. SUMMARY

Materials compatibility considerations have weighed heavily in the design iterations of fusion reactors leading toward commercial feasibility. The relatively large number of blanket-coolant combinations examined in design studies to date is in part a reflection of uncertainties in the chemical behavior of materials under the unique operating conditions of fusion energy systems. Among the more unique features of these systems are the presence of strong magnetic fields in the case of magnetically confined plasmas and the requirement to breed and contain tritium.

Much compatibility information that is useful to the selection of coolants for fusion reactors derives from fission reactor studies of water, sodium, and helium. The corrosion properties of water, although extensively researched, raise questions when applied to fusion reactors because of uncertainties as to what techniques could be used to control dissolved oxygen. Sodium is compatible with austenitic stainless steels to higher temperatures than is lithium, but it is not clear how this temperature advantage can be exploited with current blanket choices. Helium affords a relatively inert coolant system with iron- or nickel-base alloys, but the impurity level reactor-grade helium limits the use of reactive or refractory metals to temperatures below 400°C.

Much information also exists on lithium-bearing molten fluoride salts such as LiF-BeF₂, which are of interest as blanket materials. However the compatibility of such salts may be adversely affected either by highly reactive fluoride species, which are formed as the lithium transmutes to tritium, or by electrical potentials induced by MHD effects.

Most of the available compatibility information for lithium relates to refractory metals at temperatures higher than those of interest to fusion reactors. Among the principal problems to be faced in the use of lithium as a blanket or coolant is the relatively rapid dissolution of chromium and nickel from conventional alloys above 500°C. Corrosion rates of iron-base alloys may also be accelerated by magnetic field effects, which reduce the effective boundary layer thickness in the lithium at the containment surface. The addition of aluminum to lithium inhibits corrosion of nickel-bearing alloys by the formation of a thin, highly protective intermetallic compound at the lithium interface.

Because the operating conditions of fusion reactors are so unique in terms of neutron flux, magnetic fields, and tritium recycling, the extant corrosion data for breeding and

coolant materials await a very objective assessment of fusion reactor compatibility problems. To obtain data that are prototypic for fusion reactors will require an extremely complex and expensive experimental program. Such information eventually must await a firmer reactor design and a narrowing of the materials to be tested.

However, several compatibility issues are fundamental to the feasibility of conceptual reactor designs and lack even objective data.

One such issue is the lithium compatibility of Ni-Cr-Fe alloys containing more than 1% Ni and the temperature restriction on the alloys imposed by temperature-gradient mass transfer. Another feasibility issue concerns the radiation and chemical stability of molten nitrate salts in contact with blanket and first-wall materials. Compatibility with oxidizing impurities in helium and tritium levels that could be tolerated in the helium are critical in establishing the usefulness of titanium alloys as first-wall materials. Also carbon and hydrogen mass transport between vanadium and iron-base alloys in lithium is a significant factor in the temperature capability of first-wall materials in fusion reactors. Finally, the effect of strong magnetic fields on the mass transport kinetics of first-wall materials in liquid lithium and molten salts is an important consideration in setting the flow velocities and piping geometries of MCFRs. Although this latter compatibility problem is highly design specific, a preliminary assessment of the nature of magnetic field effects on dissolution kinetics is needed to gauge the feasibility of integrating the breeding blanket and coolant circuits.

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