Journal of Nuclear Materials 385 (2009) 217–222

Contents lists available at ScienceDirect

Journal of Nuclear Materials

journal homepage: www.elsevier.com/locate/jnucmat

Structural materials challenges for advanced reactor systems

P. Yvon *, F. Carré

CEA, Nuclear Energy Division, CEA Saclay, 91191 Gif/Yvette, France

ARTICLE INFO

PACS:
28.41.Qb
61.80.Hg
61.82.Bg

ABSTRACT

Key technologies for advanced nuclear systems encompass high temperature structural materials, fast neutron resistant core materials, and specific reactor and power conversion technologies (intermediate heat exchanger, turbo-machinery, high temperature electrolytic or thermo-chemical water splitting processes, etc.). The main requirements for the materials to be used in these reactor systems are dimensional stability under irradiation, whether under stress (irradiation creep or relaxation) or without stress (swelling, growth), an acceptable evolution under ageing of the mechanical properties (tensile strength, ductility, creep resistance, fracture toughness, resilience) and a good behavior in corrosive environments (reactor coolant or process fluid). Other criteria for the materials are their cost to fabricate and to assemble, and their composition could be optimized in order for instance to present low-activation (or rapid desactivation) features which facilitate maintenance and disposal. These requirements have to be met under normal operating conditions, as well as in incidental and accidental conditions. These challenging requirements imply that in most cases, the use of conventional nuclear materials is excluded, even after optimization and a new range of materials has to be developed and qualified for nuclear use. This paper gives a brief overview of various materials that are essential to establish advanced systems feasibility and performance for in pile and out of pile applications, such as ferritic/martensitic steels (9–12% Cr), nickel based alloys (Haynes 230, Inconel 617, etc.), oxide dispersion strengthened ferritic/martensitic steels, and ceramics (SiC, TiC, etc.). This article gives also an insight into the various natures of R&D needed on advanced materials, including fundamental research to investigate basic physical and chemical phenomena occurring in normal and accidental operating conditions, lab-scale tests to characterize candidate materials mechanical properties and corrosion resistance, as well as component mock-up tests on technology loops to validate potential applications while accounting for mechanical design rules and manufacturing processes. The selection, assessment and validation of materials necessitate a large number of experiments, involving rare and expensive facilities such as research reactors, hot laboratories or corrosion loops. The modelling and the codification of the behaviour of materials will always involve the use of such technological experiments, but it is of utmost importance to develop also a predictive material science. Finally, the paper stresses the benefit of prospects of multilateral collaboration to join skills and share efforts of R&D to achieve in the nuclear field breakthroughs on materials that have already been achieved over the past decades in other industry sectors (aeronautics, metallurgy, chemistry, etc.).

1. Introduction

The fast growing energy demand and concerns about climate changes require nuclear energy to play a role among other energy sources to satisfy future energy needs of mankind. Generation III light water reactors (LWRs) are anticipated to be built in large numbers to replace existing nuclear power plants or to augment the nuclear production capacity. Beyond the commercialization of best available light water reactor technologies, it is essential to start now the development of breakthrough technologies that will be needed to prepare the longer term future for nuclear power. These innovative systems include fast neutron reactors with a closed fuel cycle and high temperature reactors which could be used for process heat applications.

- Fast neutron reactors (sodium, gas or lead cooled) with a closed fuel cycle which afford making an efficient use of uranium resource (more than 80% instead of 1% at most by light water reactors which essentially consume ^{235}U) and minimizing long-lived radioactive waste, thus making nuclear energy more sustainable.
- High temperature reactors that may drive more efficient processes to generate other energy products than electricity such as hydrogen, synthetic hydrocarbon fuels from coal or biomass, or process heat for the industry, thus contributing to enlarge the range of applications of nuclear energy.

* Corresponding author. Tel.: +33 169086856; fax: +33 169085892.
E-mail address: pascal.yvon@cea.fr (P. Yvon).

0022-3115/$ - see front matter © 2008 Elsevier B.V. All rights reserved.
doi:10.1016/j.jnucmat.2008.11.026

- Other innovative systems such as thermal or fast neutron spectrum supercritical water reactors and thorium fueled molten salt reactors which may offer attractive prospects for future energy supply.
- Accelerator Driven Systems (ADS) which may contribute to transmute minor actinides in dedicated systems, in addition to transmutation that could be achieved through advanced recycling modes in fast reactor systems and
- Fusion reactors, which use elements abundant in nature, lithium and deuterium as fuel, and do not produce long-lived radioactive waste.

Several initiatives today such as the Generation IV International Forum [1] and the IAEA International Project on Innovative Nuclear Reactor (INPRO) [2,3] aim at revisiting the technologies that led to early prototypes of fast neutron and high temperature reactors, and to search for innovations that could make them progress significantly in competitiveness, safety and operability so as to prepare the development of attractive commercial nuclear systems. Fast reactors are expected to be needed around 2040 if, as anticipated today, the installed capacity of LWRs reaches an electrical power of 1300–1500 GW by 2050, and uranium prices escalate because most of the estimated resource below 130 US$ per kg (i.e. $\sim$15 megaton) is preempted by fueling needs of these reactors over a 60-year lifetime. Mixed oxide (MOX) fuelled high conversion reactors may smooth out the transition from LWRs to fast spectrum reactors while somewhat relaxing the uranium demand around the middle of the 21st century and thus postponing the need for fast spectrum reactors by a few decades. However, if their technical maturity and competitiveness arise earlier, fast neutron reactors could support the industrial implementation around 2030 of the Global Nuclear Energy Partnership [4] 's strategy that is proposed by the US-DOE to safely develop nuclear power worldwide with an adequate control of proliferation risks. This strategy is based on a practice of nuclear fuel leasing and take-back services assured by "fuel-cycle states" that would operate such fast neutron reactors to recycle nuclear materials recovered from retrieved LWR spent fuel. For either of both above applications, fast neutron reactors are being revisited along basically three tracks: innovative sodium cooled fast reactors (SFRs) that are likely to lead to prototypes around 2020–25 and be ready for industrial deployment by 2040 or earlier, and alternative technologies, gas or lead-alloy cooled fast reactors (GFRs and LFRs), that call for experimental technology demonstrators around 2020 prior to considering prototypes around 2030–35, and industrial deployment after 2050. Besides, very/high temperature reactor (V/HTR) energy products might become marketable as early as 2025, especially to oil and refinery companies that need high temperature process heat and hydrogen already today and possibly synthetic hydrocarbon fuels from coal or biomass to complement fuels from fossil origin. Lastly, feasibility and scoping studies are shared internationally on more prospective reactor types such as the supercritical water reactor (SCWR) and the molten salt reactor (MSR) to further assess crucial feasibility issues and achievable performance.

Key technologies for such innovative nuclear systems encompass high temperature structural materials, fast neutron resistant fuels and core materials, advanced fuel recycle processes with co-management of actinides, possibly including minor actinides, and specific reactor and power conversion technologies (intermediate heat exchanger, turbomachinery, high temperature electrolytic or thermo-chemical water splitting processes, etc.).

2. Materials requirements for the innovative reactor systems

The main requirements for the materials to be used in these reactor systems are the following:

- The in-core materials need to exhibit dimensional stability under irradiation, whether under stress (irradiation creep or relaxation) or without stress (swelling, growth).
- The mechanical properties of all structural materials (tensile strength, ductility, creep resistance, fracture toughness, resilience) have to remain acceptable after ageing, and
- The materials have to retain their properties in corrosive environments (reactor coolant or process fluid).

Other criteria for the materials are their costs to fabricate and to assemble, and their composition should be optimized in order for instance to present low-activation (or rapid deactivation) features which facilitate maintenance and disposal. These requirements have to be met under normal operating conditions, as well as in incidental and accidental conditions. These demands are similar in their nature to those required for the current operating commercial reactors, but are actually much more demanding, due to the specifications of the innovative systems. It can be noted that, when the operating temperature of commercial light water reactors does not exceed 625 K, the levels of temperature required here are much higher as illustrated in Fig. 1 [5] which represents a major challenge. Another tough challenge is the high irradiation doses sustained by the in-core materials. Also, it can be difficult to find materials compatible with some of the coolant or process fluids considered. The combination of high temperature, high neutron dose and environment could prove to be a major obstacle for the viability of some of the systems. Finally, the toughest demand is the lifetime expectancy, which is 60 years, where the maximum design life of reactor materials is more in the range of 30 years [6]. Table 1 summarizes the different classes of materials considered for each system.

To meet these specifications, existing commercial or near commercial materials have to be assessed and qualified. They may require optimization and in some instances new materials have to be developed. This means extensive experimental programs have to be carried out in order to characterize the ageing capacity, the corrosion resistance and the mechanical behavior before and after irradiation of base materials as well as welds. The behaviour under irradiation of a wide range of structural materials such as graphite (V/HTR and MSR), austenitic and ferritic steels (V/HTR, SFR, GFR, LFR), Ni-based alloys (SCWR), ceramics (GFR) has to be assessed... Experimental irradiations have to be carried out in order to study microstructural and dimensional evolution, but also the behaviour under stress. Corrosion tests have also to be performed in representative environments. These characterization programs are lengthy and costly. A good basic understanding of the mechanisms controlling material performance will allow an optimisation of

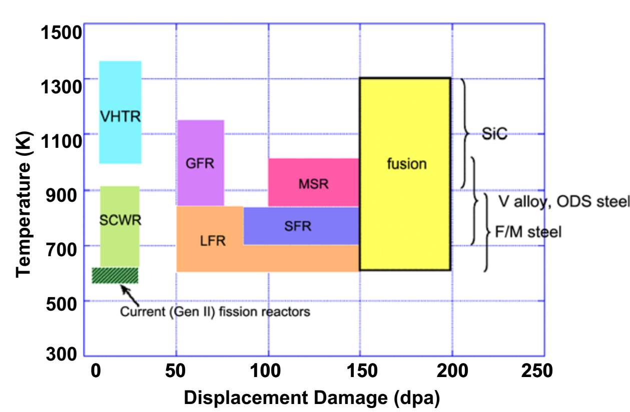

Fig. 1. Operating temperature as a function of displacement dose for different reactor systems (from [5]).

Table 1
Summary of materials considered for the different systems.

	SFR	GFR	LFR	VHTR	SCWR	MSR	Fusion	
Coolant	Liquid Na	He	Lead alloys	He	Water	Molten salt	He	Pb–17Li,
P (MPa)	0.1	7	0.1	7	24	0.1	8	0.1
T (K)	625–825	755–1125	825–1075	875–1275	555–825	775–995	575–855	755–975
Core structures	Wrapper F/M steels	Fuel & core structures	Target, Window Cladding	Core Graphite	Cladding & core structures	Core structure	First wall Blanket	
	Cladding Adv Aust & F/M steels			Control rods		Graphite	F/M steels ODS	
	F/M ODS	SiC_f/SiC composite	F/M steels ODS	C/C SiC_f/SiC	Ni based Alloys & F/M steels	Hastelloy	SiC_f/SiC	
T (K)	665–975	875–1475	625–755	875–1875	625–895	975–1075	775–900	
Displacement dose (dpa)	Cladding 200	60 = 90	Cladding ~00 dpa, ADS/ target ~100 dpa	7/25			~100 + 10 ppmHe/ dpa + 45 ppmH/dpa	
Other components		IHX or turbine Ni alloys		IHX or turbine Ni alloys				

these tests, with the extraction of maximum information. Furthermore it is of high interest to look for commonalities in-service conditions for different systems. For instance, LFR materials have benefited from research done in the frame of ADS programs and fusion materials (F/M steels, oxide dispersion strengthened alloys (ODS), SiC/SiC) are also widely considered for Generation IV systems.

In the following section, several promising candidate materials will be described in more detail with an emphasis on the benefits of crosscutting research programs. Also, the need to develop predictive modeling tools will be addressed.

3. Candidate materials for innovative reactor systems

3.1. Ferritic/martensitic steels

Ferritic/martensitic steels (9–12% Cr) are promising candidate materials for sodium or lead cooled reactors with a high temperature (<875 K) and compact primary system, as well as for the pressure vessel of high temperature gas-cooled reactors. They are also considered for core structure materials of fusion reactors. First, there is a strong push to reach high burn ups to optimize the use of resources and to minimize the waste, and the swelling of alloys currently used for fuel cladding in fast neutron reactors limits the burn up, because of geometrical constraints and also the loss of mechanical strength. This limitation could be overcome with improved materials such as ferritic/martensitic steels which exhibit a much lower swelling as shown in Fig. 2 which shows that the maximum hoop deformation of austenitic steels increases rapidly after an "incubation" period when that of F/M steels remains low even at displacement doses above 150 displacements per atom (dpa). This also shows the need to be thorough in the material characterization, since no problem seems to exist until the displacement dose reaches ~70 dpa – at that dose, a swelling mechanism gets triggered for some grades of austenitic alloys. Also, implementing low swelling materials may allow reducing coolant channels thickness, which is one of the paths to cope with coolant void reactivity coefficient, a major safety issue for SFR.

Structural reactor materials and fuel cladding materials have to be identified to withstand the combined effects of high temperatures, harsh irradiation conditions and the rather corrosive/erosive operating conditions of lead or lead-alloys coolant. A careful selection of steels meeting these conditions from the present nuclear materials database is required. Austenitic steels like AISI 316L that are used in water cooled reactors are not only susceptible to irradiation induced swelling and creep, but also show limited corrosion resistance in Pb-alloys at high temperature. Ferritic–martensitic steels have better corrosion behaviour in liquid Pb-alloys than austenitic steels. They also present a better behaviour under irradiation and thus appear to be candidate materials for fuel cladding and structures in high flux zones.

The 9% Cr martensitic steels with their Low Activation (LA) variants for fusion are foreseen for operating temperatures up to 825 K. A large set of data issued from the various Fast Neutron Reactors programmes exist for the classical 9–12% Cr martensitic steels for doses ~100 dpa in the range 675–825 K. The resistance to swelling is excellent due to the bcc crystalline structure and the high density of sinks of the martensitic microstructure. The hardening and embrittlement are negligible when irradiation occurs in the range 675–825 K. Finally, these alloys exhibit better mechanical properties, lower thermal dilation and are cheaper than alloys used previously in the intermediary circuit of sodium cooled fast reactors, which would help design a much more compact circuit and make the system more competitive economically.

3.2. Oxide dispersion strengthened steels

Oxide dispersion strengthened ferritic/martensitic steels are considered for cladding materials for high burn-up fast neutron

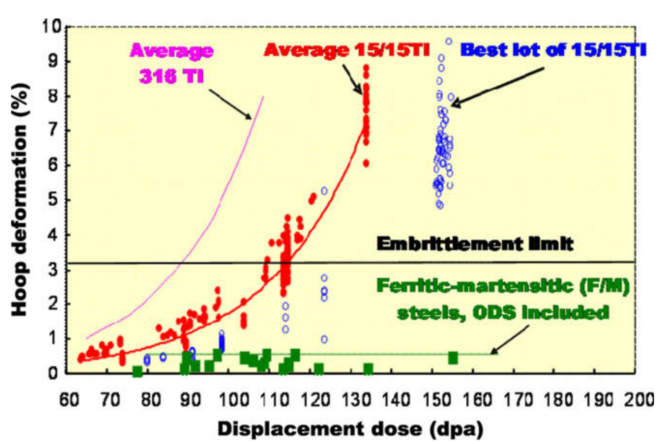

Fig. 2. Maximum hoop deformation of different grades of austenitic Phénix claddings and ferrito-martensitic materials versus dose at temperatures between 675 and 825 K.

reactor fuels. Fig. 2 shows the benefits of a ferritic/martensitic matrix with respect to swelling problems. The nanosized dispersoids of yttrium oxide give these alloys a good creep resistance at high temperatures [7]. The ODS grades currently developed in the frame of the SFR or fusion contain 9–12% Cr. However, these alloys could show some limitations in terms of internal corrosion (oxide clad reaction) and temperature (phase transition around 1075 K). Therefore, ferritic steels with ~14% Cr and more could be used up to 1175 K. Although irradiation data are scarce, the bcc crystalline structure should present an excellent resistance to swelling. The main in-service issues, in the low temperature range, remain the effect of the α/α' unmixing on the mechanical properties and, in the high operating temperature domain, the required stability of the oxide dispersion to maintain the improved creep resistance of this type of material and the absence of heavy intermetallic phase precipitation that could degrade the toughness of the cladding as illustrated in Fig. 3 [8]. Preliminary results under mixed and fast neutron spectrum show that α/α' demixing should allow this type of materials to keep reasonable ductility and fracture toughness. The under irradiation stability of the oxide dispersion is an open issue to be settled. The action of the oxide dispersoids on the irradiation creep, where climb phenomena are predominant, remains to be understood. Other types of strengthening precipitates may also be considered (e.g. carbides and nitrides).

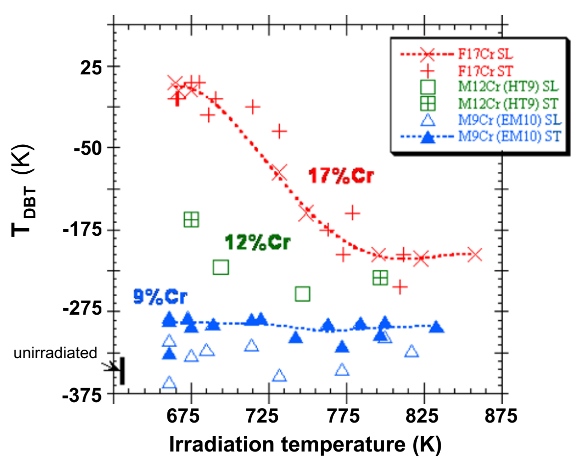

Fig. 3. Ductile to brittle transition temperature (T_{DBT}) for F/M steels with different Cr contents as a function of irradiation temperature (from [8]).

3.3. Ceramic materials

Ceramic materials are needed for very high temperature components (>1275 K) such as heat exchangers and thermal insulations in the primary system, as well as core components such as control rod sheath (V/HTR and GFR) and fuel constituents (GFR as shown in Fig. 4). A major effort of research is being invested in developing less brittle ceramics forms such as composite ceramics, or nano-structured plastic ceramics. SiC ceramics, that have been extensively investigated for Fusion applications [9], are the major focus of R&D today but other carbides (TiC, ZrC) or nitride (TiN, ZrN, etc.) are also currently considered in screening tests [10,11]. Ceramics are also needed for parts of power conversion systems such as very high temperature gas turbine blades or intermediate heat exchangers to decompose sulfuric acid (>1125 K) as high temperature step of several thermo-chemical water splitting processes (iodine–sulfur [12], Westinghouse hybrid [13], etc.).

The main goals of the studies on non metallic materials such as graphite or ceramic matrix composites (C/C, C/SiC, SiC/SiC) are to select and characterize various core materials for structural applications (reflectors, fuel structures, control rod cladding, core's support plates) at high temperatures ($T \sim 1375$ K up to 1875 K in accidental conditions) under irradiation and impure helium environment.

C/C composites are planned to be used for control rod cladding (start up/shut down control rods and operating control rods) and core support plates because of their high specific resistance and their good mechanical behaviour at temperatures above 1375 K. Unfortunately these materials are sensitive to oxidation at such temperatures and their behaviour under irradiation is badly known and this is particularly critical for operating control rods. Some industrial solutions exist to improve the oxidation resistance (protective coating, SiC final or full impregnation, self-healing matrix, SiC/SiC) but irradiation behaviour might be a real problem (risk of swelling, drop of thermal conductivity and mechanical properties, damage, failure).

Fig. 4. SiC/SiC fuel pin or fuel plate for GFR.

The fibre architecture is generally specific of the component in order the fibres to bear the thermal–mechanical loading, and the grades used for fusion might not suit V/HTR applications. Indeed, the fibre nature, the fibre architecture (2D, 3D, etc.), the matrix nature, the heat treatment and the geometry of the components have to be adjusted to meet the specific requirements of control rods. At last, composite materials are inhomogeneous and most of the time anisotropic. Moreover, there is a wide dispersion during mechanical characterization, which necessitates a large quantity of tests to qualify a composite material. Acceptable values of fracture toughness $\sim$25 MPa $m^{1/2}$ are obtained via tailored inter-phases between the fibres and the matrix. The tensile property and dimension stability of this type of composite, cubic and stoichiometric, has been proven satisfactory only up to moderate doses around 10 dpa under mixed neutron spectrum. The main issues are (i) long term stability of dimension and physical properties, (ii) irradiation detrimental effect on the inter-phase and its capability of deviating cracks and thus providing reasonable fracture toughness, (iii) required higher creep strength of the fibre to bear the thermal–mechanical loading in long term service under high temperature and neutron flux, (iv) type of mechanical damage under irradiation and creep. The behaviour of these materials under coupled irradiation and mechanical stress needs to be assessed. This is a major challenge for fuel development where containment high performance and evolution under irradiation have to be demonstrated.

This rapid survey is pointing out that there is a real need for irradiation data (on composite materials and on fibres themselves) for better dimensioning the composite materials. The oxidation behaviour has to be evaluated too to ensure the integrity of the control rods in case of accidental air ingress. These two points appear to be the most important ones. Data on thermo-mechanical properties are also needed on each grade selected to feed data bases. Testing components as representative as possible of the final structure is necessary too. Indeed, results obtained on flat samples might be different from results obtained on tubes. The problem of tested sample representativeness will be of real importance for irradiation experiments where the size and the number of samples are limited. At last, modelling of the composite behaviour will be essential for assessing the lifetime of components.

3.4. Nickel based alloys

Nickel based alloys (Haynes 230, Inconel 617, etc.) are the reference candidate materials for high temperature gas-cooled primary system components such as the intermediate heat exchanger, and oxide dispersion strengthened grades are considered to match requirements for higher temperature service conditions (intermediate heat exchanger, gas turbine blades, etc.).

New concepts of nuclear reactors, based on gas coolant systems at very high temperature (V/HTR) with a gas turbine are theoretically able to deliver higher efficiency than conventional steam cycle, by means of using very high efficiency components, as helium/helium recuperator heat exchangers or helium/gas mixture intermediate heat exchangers. These components should work under very harsh conditions (very high temperature, 1075–1275 K, and high pressure, 5–8 MPa). According to a recent investigation, no proven industrial technology could be directly used. The main topic of these heat exchangers concerns the use of nickel base high temperature materials and more precisely the coupling between ideal geometries coming from the optimised design, and the ability of the material to be formed and assembled according to this requirement.

The most critical components of the turbomachinery in a VHTR direct cycle are the first stages of the turbine disks. In order to achieve a high efficiency, it is recommended to limit the cooling of the turbine. For the blades, materials such as directionally solidified superalloys or single crystals should meet the specifications. The only investigations launched so far are related to corrosion. For the disk, no commercial materials satisfy the specifications of this gas turbine (maintenance intervals of 60000 h, unusual impure helium environment, maximum temperature of the disk expected between 975 and 1025 K, disk size of 1.5 m). Among the materials issued from the aircraft and the land-based gas turbine industries, the Ni-based superalloy Udimet 720 constitutes the best candidate. This alloy is elaborated either through a cast and wrought (C&W) process route either through a powder metallurgy one. For the latter, the powder is first consolidated in a Hot Isostatic Pressure (HIP) facility, optionally followed by a forging operation. The C&W process leads to a high creep resistant grade, but the maximum size of a forged disk ever manufactured is less than 1 m. The developments required to manufacture larger disk have to be investigated. The HIP process allows for manufacturing of large disks, but usually leads to intermediate creep resistance due to the small grain size generated by this process.

The material for the primary circuit must exhibit very good thermal stability for long operating time, moderate creep strength, well established metal working and welding techniques. The candidates will thus be selected within the class of Ni-base solid solution strengthened superalloys. During the past, exhaustive work has been performed on Inconel 617 which was the candidate material of the German team [14,15] and on Hastelloy XR (it is an adapted grade of Hastelloy X for VHTR application) which was the chosen material for the Japanese team [16]. Many results have been reported in the literature on both materials. Inconel 617 exhibits the best creep properties, but suffers from a quite poor corrosion resistance (non-protective oxide film leading to internal oxidation and decarburization). Moreover its high Co content may give rise to potential radioactive contamination problem. Hastelloy XR exhibits very good stability in VHTR environment, but does not perform as well in terms of creep resistance. CEA has selected Haynes 230 as a very promising candidate material for this application. Indeed, this material has been developed to withstand aggressive environment, and should exhibit good creep properties. Experimental programs have been launched to qualify alloy 230 for this specific application. It includes, long term thermal exposure treatments, mechanical characterization (tensile, impact toughness, creep under air and vacuum) and corrosion studies (static exposure tests and creep tests under impure atmosphere). Both the bulk material and relevant welds are investigated.

The evolution of the microstructure under thermal ageing is accurately investigated in order to be able to predict the behavior of the material under very long service period. The first results obtained at 1025 K have revealed the formation of carbides either close to the grain boundaries (with a typical lamellar structure) either close to the former coarse W-rich carbides. This change in the microstructure leads to a limited drop of the mechanical properties of the material. It seems that after 1000 h at 1025 K the material has reached a stable structure. Similar investigations at 1125 K are underway. First results of the creep tests under air and vacuum have indicated a creep resistance similar to that of Inconel 617.

Different ageing treatments have been characterized to highlight the influence of the microstructure on the creep resistance of the material. Indeed, it has been established that at high stresses, the intra-granular microstructure (i.e. the γ' populations) is responsible for the creep strength whereas at low stress and high temperature, the grain size is the controlling factor. It is thus recommended to coarsen the grain size to increase the creep resistance. However, during super-solvus ageing treatment, the grain boundaries are pinned by the small precipitates decorating the prior particles such that the austenitic grain size is limited to the powder granulometry. To overcome this limitation, coarser grain size material (~100 μm), for which the austenitic grain is larger than the former powder particles, can be developed.

4. Modelling

As mentioned above, the selection of materials, the assessment of their properties and their qualification necessitate a large number of experiments, involving rare and expensive facilities such as research reactors, hot laboratories or corrosion loops. The modelling and the codification of the behaviour of materials will always involve the use of such technological experiments, but it is of utmost importance to develop a predictive material science: in most cases the experimental database does not cover the whole range of conditions needed (temperature, time, neutron spectrum and flux) and it is essential to develop robust physical models in order to extrapolate with confidence outside the experimentally known domain. Furthermore, modelling and simulation can give access to basic information hardly obtainable by global experiments (standard irradiation tests with their post-irradiation examinations), e.g. on the dynamics of impurities and defects under irradiation. This understanding of the mechanisms underlying the material behaviour is critical to identify and control the most relevant parameters (for instance little is known on the actual structure of oxide nano-clusters in ODS steels [17], their coherence with the iron matrix, the mechanisms responsible for their impact on the overall potential behaviour in-service conditions). This will also prove a powerful tool to optimize existing materials or develop new materials.

Modelling must make the best use of the continuously increasing computing capabilities, but must also be supported by dedicated experiments. Our understanding of radiation damage at the atomic scale now allows simulating radiation induced microstructures and dislocation dynamics in volumes that are of the same order of magnitude as those that can be irradiated (with ion beams of a few MeV), characterised physically and chemically (by transition electron microscopy, X-rays, atom probes, ion beam analysis, etc.), and characterised mechanically (nano-indentation, TEM testing, etc.). For example, coupling multi-ion beams irradiation with modern characterisation like in the JANNUS facility developed by CEA and CNRS, will provide a way to validate the atomic scale modelling and build predictive tools for radiation effects up to high dose and content of transmutation products. These experimental and theoretical tools will be the basis for developing materials more resistant to radiation effects, designing optimised test programmes in dedicated neutron sources (material testing reactors, IFMIF), extrapolating their results to the wider range of in-service conditions of the innovative reactor systems (both GenIV and fusion), and finally giving higher confidence in the data involved in the licensing processes.

The physical modelling based on the theory and computation of atomic cohesive forces that control thermodynamics, kinetics and mechanics has already been undertaken for LWR structural materials for instance in the frame of the integrated project PERFECT of the 6th Framework Program of the EU [18], and it is important to capitalize on such experience, since the generic nature of this modelling makes it applicable to generation IV and fusion. There is a certain maturity of the modelling in the field of thermodynamics and kinetics in metals and alloys, and it is possible for instance to predict with some confidence the evolution of microstructure under irradiation [19]. This level of physical robustness has now to be extended to mechanical modelling by coupling microstructure and mechanics (plasticity – creep – fracture), and to other materials (such as nano-structured alloys, ceramics and composite materials).

5. Conclusions

The development of innovative nuclear systems requires extensive research to find, qualify and codify materials able to withstand the extreme demands in terms of temperature, high neutron flux, corrosive environment and lifetime expectancy. In most cases, the viability of these concepts relies on the capacity of obtaining such materials.

The development and implementation of advanced metallic alloys and ceramic composites will need breakthroughs in material science, from process development (material fabrication, assembling, etc.) to performance assessment (behaviour under coupled temperature, mechanical stress and irradiation). These challenges require comprehensive tests and in-depth investigations of structural materials.

The extensive characterization of candidate materials will require a joint effort from the international community, and these efforts have to be mutualised between the different systems as synergies have already proven to be helpful. Finally, besides the experimental testing, a strong emphasis should be placed on developing a predictive material science and robust modelling, based on the physics of materials at the atomic and grain scale.

References

[1] <ttp://geniv.ne.doe.gov>.
[2] J. Kupitz, in: International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), ICAPP03, 2003, p. 3110.
[3] IAEA International Project on Innovative Nuclear Reactor (INPRO), Final report of phase 1, 2006.
[4] <http://www.gnep.energy.gov>.
[5] S.J. Zinkle, in: NEA Workshop Proceedings, Karlsruhe, 4–6 June 2007.
[6] W.R. Corwin, Nucl. Eng. Technol. 38 (2006) 591.
[7] A. Alamo, V. Lambard, X. Averty, M.H. Mathon, J. Nucl. Mater. 329–333 (2004) 333.
[8] J.L. Séran, A. Alamo, A. Maillard, H. Touron, J.C. Brachet, P. Dubuisson, O. Rabouille, J. Nucl. Mater. 212–215 (1994) 588.
[9] L.L. Snead et al., J. Nucl. Mater. 283–287 (2000) 551.
[10] A. Gusev, A. Remple, G. Shveikin, Doklady Phys. Chem. 357 (4–6) (1997) 373.
[11] D. Gosset, M. Dollé, D. Siméone, G. Baldinozzi, L. Thomé, J. Nucl. Mater. 373 (2008) 123.
[12] D. O'Keefe, C. Allen, G. Besenbruch, L. Brown, G. Norman, R. Sharp, et al., Int J. Hydrogen Energy 7 (1982) 381.
[13] L.E. Brecher, S. Spewock, C.J. Warde, Int J. Hydrogen Energy 2 (1977) 7.
[14] .F. Schubert et al., Nucl. Eng. Des. 78 (1984) 251.
[15] H. Breitling et al., in: Proceedings of a Symposium on High Temperature Metallic Materials for Gas-Cooled Reactors, Cracow, IAEA IWGGCR-18, 1988, p. 96.
[16] Y. Kurata, H. Nakajima, J. Nucl. Sci. Technol. 32 (1995) 539;
Y. Kurata, H. Nakajima, J. Nucl. Mater. 228 (1996) 176.
[17] C. Degueldre, S. Conradson, W. Hoffelner, Comput. Mater. Sci. 33 (2005) 3.
[18] <http://www.fp6perfect.net>.
[19] R.A. Enrique, P. Bellon, Phys. Rev. Lett. 84 (2000) 2886.