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European cross-cutting research on structural materials for Generation IV and transmutation systems

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ABSTRACT

It has been internationally recognized that materials science and materials development are key issues for the implementation of innovative reactor systems such as those defined in the framework of the Generation IV and advanced fuel cycle initiatives. In Europe, materials studies are considered within the Strategic Research Agenda of the Sustainable Nuclear Energy Technology Platform. Moreover, the European Commission has recently launched a 7th Framework Programme Research Project, named 'Generation IV and Transmutation Materials', that has the objective of addressing materials issues which are cross-cutting for more than one type of innovative reactor systems. The present work has been prepared with the aim of describing the rationale, the objectives, the work plan and the expected results of this research project.

1. Introduction

Materials science and materials development are important topics to support the development of Generation IV reactors and advanced fuel cycle systems, as well as to optimise the operation of 2nd and 3rd Generation Light Water Reactors (LWRs). As far as the European framework is concerned, recently a number of European nuclear stakeholders have issued a vision report to establish the Sustainable Nuclear Energy Technology Platform (SNE-TP) [1]. The SNE-TP is preparing the European Strategic Research Agenda (SRA) with the objectives of optimising operating and next generation LWRs, of preparing the development of sustainable nuclear systems for the future, including advanced fuel cycles, and of widening the range of nuclear energy applications. The preliminary roadmaps that account for the aforementioned objectives have identified relevant milestones for the years 2012, 2020 and 2040. These can be summarised as follows. By 2012: (1) confirmation of Sodium Fast Reactor (SFR) design options and selection of a second type of fast neutron system of importance to Europe (this second type of fast system could be of the Lead Fast Reactor (LFR) type, sub-critical Accelerator Driven System (ADS) or Gas Fast Reactor (GFR) type); (2) viability of Super Critical Water Reactor (SCWR); (3) confirmation of key technologies for the Very High Temperature Reactor (VHTR). Furthermore, it is foreseen that by 2020 the start-up of the SFR and the second type of fast neutron system will occur, as well as the construction of a VHTR and the related demonstration for cogeneration applications. Finally, in the time-frame 2020–2040, further R&D will be needed to design and optimise full-scale systems in order to build a first-of-a-kind fast reactor and start commercial deployment.

Although significant differences exist among the different reactor concepts under consideration in Europe and despite the fact that at present no definitive design has been established for any of them, the operating conditions envisaged for those systems are quite demanding and they will impact the performance of the structural materials. In fact, for the declared objectives of increasing efficiency and enhancing economy of the nuclear systems, high operating temperatures and high burn-ups are important goals of the process engineering. However, the safety and the feasibility of most of these nuclear reactor concepts and their optimisation will depend crucially on the capability of the chosen structural materials to withstand the expected operating conditions under specific thermal-hydraulics conditions (in the presence of one of the several coolants presently under investigation).

In this context, the European Commission has launched in its 7th Euratom Framework Programme, a research project to qualify commercially available materials and for the longer term to develop and qualify new materials and corrosion protection barriers. This project has been named 'Generation IV and Transmutation Materials' (GETMAT) [2].

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2. Innovative nuclear systems and material issues

In order to define the rationale and objectives of the GETMAT research programme, which is cross-cutting for more than one reactor concept, an analysis of European and national projects and programmes aimed at designing innovative nuclear systems has been performed [3,4].

Key materials issues identified for the development of GEN IV and transmutation systems are those related to core components, which are subjected to the most severe in-service conditions. Besides the fuel itself, other structures of the fuel subassemblies, such as fuel cladding and wrapper tubes, must operate under high levels of radiation damage, high temperature and coolant effects (sodium, Pb or Pb–Bi eutectic, gas or supercritical water).

The operating conditions foreseen for the innovative reactor systems are summarised in Table 1. As shown in this table two classes of materials have been identified for the core components of most systems, i.e. the high-Cr Ferritic/Martensitic (F/M) steels and Oxide Dispersion Strengthened (ODS) Fe–Cr alloys. Synergies with the fusion technology materials development programme have also been identified e.g. in the areas of ODS fabrication procedures, irradiation programmes and multi-scale modelling. Typical low activation F/M and ODS steels developed for fusion will be included for comparison in the GETMAT evaluation programme.

Despite the existing sizeable experience on the high-Cr F/M steels [5–7], further data are needed to qualify their use in the service conditions planned for the nuclear systems. On the other hand, the use of ODS alloys for nuclear application is a nearly unexplored domain, where knowledge-improvement is considered an essential task.

The focus of the GETMAT project is on the development and characterisation of ODS and high-Cr F/M steels, their welding/joining and their qualification, in terms of mechanical and corrosion resistance in appropriate conditions. Moreover, an extensive Post Irradiation Examinations (PIE) programme, taking advantage of ongoing irradiation experiments (see e.g. [8,9]) will contribute to the assessment of the two classes of materials. Finally, an important part of the GETMAT project is dedicated to the fundamental understanding of the physical mechanisms that are the basis of the materials behaviour under irradiation through multi-scale modelling activities and experimental validation.

3. Objectives and structure of the GETMAT Project

As mentioned, the objective of the GETMAT project is to contribute to the selection and qualification of structural materials for the innovative reactor systems considered in the framework of Generation IV and advance fuel cycle initiatives. The GETMAT work programme has been defined in order to:

- Improve and extend the qualification of high-Cr (9–12 Cr) F/M steels. The reference F/M steel considered for several reactor systems is the modified 9Cr–1Mo ‘T91’ steel. However, other F/M steels, such as e.g. the ‘EUROFER’ and ‘T92’ steels, will be also included in this activity. The main area addressed is a wide PIE programme of relevant irradiation experiments ongoing in Europe.
- Develop and characterise (in terms of high temperature strength and resistance to corrosion) Fe–Cr ODS alloys. Two ODS alloys composition and two fabrication routes will be investigated.
- Assess and rank for the technological application of different joining and welding techniques relevant to both ODS and F/M steels.
- Develop and qualify corrosion protection barriers, in terms of corrosion and irradiation resistance. Emphasis will be put on defining a standard qualification method.
- Develop models describing radiation damage effects in Fe–Cr alloys (used as models for F/M steels) and perform relevant model experiments.

The GETMAT work programme has been defined within the four work-packages (WPs):

1. Metallurgical behaviour of ODS and F/M steels, their joining and welding;
2. Corrosion and mechanical behaviour;
3. Irradiation behaviour; and
4. Multi-scale modelling and model experiments.

In the next sections the content of the work programme defined for these four WPs is summarised.

3.1. Metallurgical behaviour of ODS and F/M steels, their joining and welding

3.1.1. Materials selection and procurement

In the past European research programmes devoted to materials studies for fast reactor systems, as e.g. the EFR project [10], austenitic steels have been considered the candidate materials for the fuel cladding, the vessel and the primary circuits. On the other hand, F/M steels were the reference choices for the wrapper

Table 1
Operating conditions of innovative nuclear systems [4].

	SFR	GFR	LFR/ADS	VHTR	SCWR	MSR	Fusion
Coolant pressure	Liquid Na few bars	He 70 bar	Pb alloys few bars	He 70 bar	Water 24 MPa	Molten salts	He/Pb–17Li 80/few
Core structures	Wrapper F/M steel Cladding AFMA F/M ODS	Fuel and core structures SiCf–SiC composite Core components ODS steels	Target and cladding F/M steels ODS	Core graphite Control rods C/C SiC/SiC	Cladding and core structures Ni-based alloys and F/M steels ODS steels	Core structures Graphite Hastelloy	First wall and blanket F/M steels ODS SiCf–SiC
Temperature °C	390–700	600–1200	350–480	600–1600	350–620	700–800	500–700
Dose	Cladding 200 dpa	60–90 dpa	Cladding ∼100 dpa ADS/Target ∼ 100 dpa	7–25 dpa	∼20 dpa	–	∼100 dpa +10 ppm He/dpa +45 ppm H/dpa
Other components	–	IHX or turbine Ni alloys	–	IHX or turbine Ni alloys RPV F/M steels	–	–	–

tube and later on they were considered for the primary systems as well [10].

Today, high-Cr F/M steels are considered the reference structural materials for the components of the primary systems (as proposed for the Japanese SFR [11]), for the cladding (as proposed for the heavy liquid metal cooled systems [12,13]) and for the vessel (as proposed for the Gas Cooled Systems [14]). The preference for F/M steels in all these applications is mainly related to the better thermal properties (higher thermal conductivity and lower thermal expansion), better dimensional stability under irradiation [7] and better compatibility to heavy liquid metals [9], when compared to austenitic steels.

However, the mechanical strength of the high-Cr F/M steels is limited to temperatures of about 550 °C [7] and, on the other hand, the use of austenitic steels (as e.g. cladding material) would limit the desired high burn-up range, due to their low resistance to radiation induced swelling [7]. These are the main reasons which have motivated the materials science community to study and develop new structural materials which could provide enhanced efficiency of the innovative reactor concepts. Within the GETMAT project, priority has been given to the development and study of Fe–Cr Oxide Dispersion Strengthened (ODS) alloys, since these show favourable properties for high temperature and high burn-up applications. Indeed, the high density of finely dispersed oxide particles of nanometer scale, that act as obstacles for moving dislocations, improve the high temperature strength of ODS materials. Therefore, ODS alloys would allow increase of the operating temperature above 550 °C. In contrast to austenitic steels and Ni-base alloys, the ODS steels do not show high temperature helium embrittlement or irradiation induced swelling [15]. Results of irradiation experiments up to 35 dpa at 330 °C indicate good irradiation resistance, with acceptable irradiation hardening and little loss of ductility [16].

Within GETMAT the fabrication of ODS-ferritic and ODS-F/M steel is an important challenge to be addressed. In particular, two fabrication routes, the powder metallurgy using a mechanical alloying procedure [17], and an alternative method making use of continuous casting technology, combined with specific thermo-mechanical alloying [18], will be explored. The advantage of the latter technology is the expected lower production cost than the powder metallurgy technique and the possibility of fabricating larger batch sizes. In summary, three batches of ODS alloys will be produced by the two techniques (see Table 2) and analysed within the GETMAT project.

3.1.2. Joining and welding technologies

The investigation of different weld techniques for ODS and F/M steels is a key issue for the different technological applications envisaged for these materials (e.g. welding of cladding tubes). The weld technologies that will be investigated in the GETMAT project are the fusion welding techniques such as Electron Beam (EB) and Tungsten Inert Gas (TIG). Welds of F/M–F/M and F/M – ODS alloys made by these techniques will be studied. For the ODS–ODS welds, the solid state techniques of Diffusion Bonding, Explosive, Friction Stir (FSW) and Electro Magnetic Pulse (EMP) will be investigated.

In contrast to austenitic steels, welded joints of F/M steels need a post-weld thermal treatment after applying EB or TIG processes to avoid hardening and embrittlement, due to martensite formation in the fusion zone, and softening in the vicinity of the heat affected zones. To improve the joint performance, the same heat treatments usually applied to the base metal, i.e. a normalisation or austenitization (typically at around 1000 °C) followed by a tempering below 800 °C are performed. To improve these welding techniques, an investigation will be performed to evaluate the best and simplest post-weld heat treatment conditions that can be applied for each process, aiming at better design criteria and increased cost efficiency during the design phase of future applications. In addition, EB and TIG are unlikely to be successfully used for ODS–ODS welding. Therefore, ODS-to-ODS solid state welding technologies will be investigated. In particular, EMP and explosive welding may be acceptable for cladding welds and diffusion bonding and FSW for joining components.

The outcome of the investigations of welding and joining techniques will be the improvement and assessment of the different techniques and a ranking of these techniques with respect to quality of the weld and their technological application (e.g. for cladding, components, etc.).

Table 2
ODS alloys produced and analysed within GETMAT.

Production technique	ODS alloy composition
Casting technology	Fe–9Cr-ODS
Powder metallurgy	Fe–9Cr-ODS
Powder metallurgy	Fe–14Cr-ODS

3.1.3. High temperature characterisation

The ODS alloys listed in Table 2 and weld samples made with the technologies discussed in the previous section will be tested for their high temperature strength. The test matrix has been defined and is shown in Table 3. These tests will evaluate the creep and fatigue resistance of these materials in an inert environment.

3.2. Corrosion and mechanical behaviour

The compatibility of materials with coolants, such as liquid metal (Pb), super critical water (SCW) or gas (He), is addressed through corrosion/erosion tests, corrosion barrier development and mechanical behaviour evaluation. Particular emphasis will be on chemical quality of the coolants, since it has been shown that impurities such as oxygen can affect the corrosion behaviour of the structural materials [19–21] in all three fluids.

3.2.1. Lead cooled systems

Materials compatibility with heavy liquid metals (HLM), such as Pb and Pb–Bi Eutectic (LBE), has been extensively studied [22]. The tests performed in stagnant HLM have shown that the chemical composition of the liquid metal (mainly related to the oxygen activity) and the temperature have an important role on the corrosion mechanisms of F/M and austenitic steels. For instance, it has been demonstrated that, generally, in the low temperature range, e.g. below 450 °C, and with an adequate oxygen activity in the liquid metal, both types of steels form an oxide layer which behaves as a corrosion barrier [9].

However, in the higher temperature range, i.e. above ~500 °C, corrosion protection through the oxide scales seems to fail. Indeed, a mixed corrosion mechanism has been observed, where both oxide scales formation and dissolution of the steel elements occurred [9]. However, in this high temperature range, it has been demonstrated that the corrosion resistance of the structural materials can be enhanced by coating the steel with FeAl alloys [9].

Ongoing experiments performed in flowing HLM (mostly LBE) [19], confirm that the corrosion mechanism of the steels depends on the oxygen content in the HLM. Indeed, at relatively low oxygen concentration the corrosion mechanism changes from oxidation to dissolution of the steel elements. Moreover, relationships between oxidation rate, flow velocity, temperature and stress conditions of the structural material have been observed [19,30].

Several mechanical tests (e.g. tensile, fatigue, slow strain rate, etc.) in HLM have also been performed [22]. The results provide evidence that, if a corrosion protection barrier is not present on

Table 3
High temperature tests on ODS materials.

Materials	9Cr–ODS alloys and welds	9Cr–ODS, 14Cr–ODS	ODS (14Cr), ODS (SCK)	14Cr ODS
Type of investigation	Creep, LCF, microstructure	Creep and creep-fatigue tests, microstructure	Small sample tests (indenter, punch, in-beam creep), microstructure	Ageing treatment at (750 and 850 °C) and tensile Charpy, and small punch test
Test-temperature °C	550, 650	600, 750	RT, 600, 750	RT, 750, 850
Test atmosphere	Inert gas	Inert gas	Inert gas, air	Vacuum, inert gas

the steel (meaning that a good contact between steel surface and liquid metal occurs) and if surface cracks (which can be considered as sites of localized concentration of stress) are present, the mechanical behaviour of F/M steel can be affected [23,24].

The materials performance assessment in HLM has focused on F/M and austenitic steels and the experiments have been conducted mainly in LBE. Within the GETMAT project, the experimental campaign will be extended to the characterisation of the ODS alloys and their welded variants in Pb and for temperatures up to 750 °C.

Moreover, creep resistance to rupture and fretting behaviour (simulating the flow induced vibration of a fuel bundle) in Pb will be investigated on F/M steels and possibly on ODS alloys.

3.2.2. Gas-cooled systems

Gas-cooled systems studied in the framework of Generation IV are the GFR and the VHTR. However, European materials research programmes for the gas cooled reactors have focussed on the VHTR concept [14,25,26].

Compatibility tests performed in gas have shown that the material behaviour strongly depends on the coolant impurity levels, temperature and flow rate. In these studies, the quality and quantity of impurities evaluated in the He were related to the VHTR system, where the identified source of impurities are the out-gassing of absorbed species from the permanent components and fuel elements, from in-leak of air during maintenance and refuelling and from operating leakages [21].

In general, it was observed that the compatibility of metallic materials with He gas depends on the carbon and oxygen activities in He [27]. Observed corrosion mechanisms are oxidation, carburisation or decarburisation, depending on the gas composition and the materials.

ODS alloys were hardly ever evaluated for their use in gas-cooled reactors. Some screening tests have been performed on these materials in the framework of the HTR development programmes in the 1970s and 1980s, but the related open literature is limited to a small number of poorly documented reports and articles.

Therefore, corrosion tests will be carried out on the GETMAT ODS alloys, in the temperature range 750 to 900 °C, taking into account the expected chemical composition of the He in the proposed GFR concept [25].

3.2.3. Supercritical-water-cooled systems

Above the thermodynamic critical point (374.2 °C and 22.1 MPa), SCW acts as a dense gas, whose properties, including density, ion products and dielectric constant, can be tuned by adjusting the temperature and pressure [28]. SCW has both liquid-like and gas-like characteristics, with good heat transport properties, acting essentially as a non polar dense gas with solvation properties approaching those of a low-polarity organic fluid. In these conditions, it can dissolve gases like oxygen to complete miscibility.

The main materials issues to be addressed concerning compatibility with SCW are the resistance to oxidation/corrosion and stress corrosion cracking [20,28,29]. However, these issues strongly depend on the SCW chemistry and the control of radiolysis gases that can be produced under irradiation [28]. In particular the ability of SCW to dissolve oxygen has to be taken into account when performing compatibility tests.

F/M and austenitic steels are candidate structural materials for SCW systems and have been extensively tested. A recent review of corrosion tests performed in SCW [20] has shown that 9Cr and 12Cr F/M steels exposed to SCW with different oxygen content exhibit double layer oxidation, similar to what has been observed in gaseous environments. The oxidation behaviour of both steels is comparable when the oxygen content in the SCW is less than 25 ppb. However, by increasing the oxygen content to 2 or 8 ppm a large scatter of experimental data is observed and the 12Cr F/M steel apparently shows less oxidation resistance.

Austenitic steels have also been tested in SCW and, similarly to the F/M steels, a double oxide layer has been observed. However, the oxide layers were thinner and they exhibited an increasing porosity with increasing test temperature.

For ODS alloys tested in SCW, data available in the open literature are scarce. These few data show that ODS alloys have a higher oxidation resistance than F/M steels, and this resistance increases with increasing Cr content.

The review of stress corrosion cracking (SCC) tests in SCW [20] showed that F/M steels seem to be resistant to SCC in SCW conditions at 400 and 500 °C and oxygen content up to 300 ppb. On the other hand, the test results of austenitic steels to SCC showed a possible relationship between the SCW dielectric constant (thus the density) and the susceptibility to SCC. In particular, by increasing the dielectric constant the fracture mode of the austenitic steels changes from ductile to inter-granular and trans-granular.

The understanding of the SCW chemistry effect on oxidation and the SCC resistance of structural materials is still an open issue, where in particular the effect of dissolved oxygen and dissolved hydrogen (hydrogen is usually used to mitigate radiolysis effects) needs to be better investigated. Moreover, a wider database is needed to assess the ODS alloys for SCWR applications. Therefore, corrosion and SCC tests will be performed on the GETMAT–ODS alloys in chemical controlled SCW conditions, at 500 and 600 °C. The experimental results will contribute to the materials selection and qualification for SCWR.

3.2.4. Corrosion protection barriers

For the compatibility of materials with the different coolants under consideration, it has been shown that, for specific thermo-chemical and thermal-hydraulics conditions, oxide scales can grow on the structural materials. In particular, the ability of these oxide scales to protect the structural materials against corrosion attack and eventual degradation of the mechanical properties has been widely studied for HLM systems (see e.g. [22]). However, as already discussed, the protection provided by the oxide scale seems to be less effective in high temperature ranges.

Moreover, recently it has been shown that, for components in HLM systems such as e.g. cladding and heat exchanger, too thick oxide scales can change the thermal conductivity of the materials, with consequences on the heat transfer of the system and on the thermo-mechanical resistance of the steel [31].

Evaluation and understanding of the parameters affecting the oxidation rate could allow an optimal range of parameters to be defined, to keep the oxide scale thickness at an acceptable level. Also, parallel research on corrosion protection will develop methods to protect components such as the cladding from corrosion/oxidation. A promising method under investigation for several years consists in spraying Fe, Al based powders on the steel surface and treating the coating with the GESA (Gepulste Elektronen Strahl Anlage) facility [32]. Corrosion tests performed on GESA treated samples in flowing HLM up to 600 °C have confirmed the effectiveness of this method [30,32].

However, it is evident that the composition of the surface layer and in particular the Al content needs to be carefully controlled in order to assure a long-term corrosion protection capability. As the next step, the composition control, and the development of a qualification method for those surface layers, will be developed.

3.3. Irradiation behaviour

A crucial point is of course the assessment and validation of the material behaviour under irradiation. In order to comply with the objectives of the project within the assigned time-frame and to account for the shut down in 2009 of the unique European fast neutron irradiation facility (Phénix), it is necessary to fully exploit ongoing irradiation programmes. As a matter of fact the combination of PIEs of different ongoing irradiation experiments will allow a wide database to be produced, complementing the experiments performed in previous programmes (see e.g. Ref. [9]). Irradiation in a fast neutron spectrum reactor (MATRIX), under proton–neutron mixed spectrum in a spallation environment (STIP) and under combined effects of neutron irradiation fields and heavy liquid metal coolant (ASTIR, IBIS, LEXURII and MEGAPIE) are considered, see Fig. 1.

A brief overview of the available data on irradiation effects and on synergistic effects of irradiation while the material is in contact with heavy liquid metal coolant is given below, in order to highlight the progress expected within the present project.

3.3.1. Irradiation effects in a fast neutron spectrum and in a spallation environment

For conventional F/M steels, a significant amount of data exists in the literature on irradiation effects of fast neutrons on dimensional stability (swelling) and mechanical properties (mainly tensile, impact and irradiation creep properties, as well as some toughness data) [6,7]. These data were generated as part of the fast breeder programmes in Europe and the US in the 1970s–1990s. The focus at that time was on materials such as the non-stabilized 9Cr–1Mo (EM10 steel, candidate material in France for wrapper tubes applications) or the 12Cr–HT9 steel. Only few data were obtained in these former irradiation programmes on the present reference material, the modified 9Cr–1Mo T91 steel.

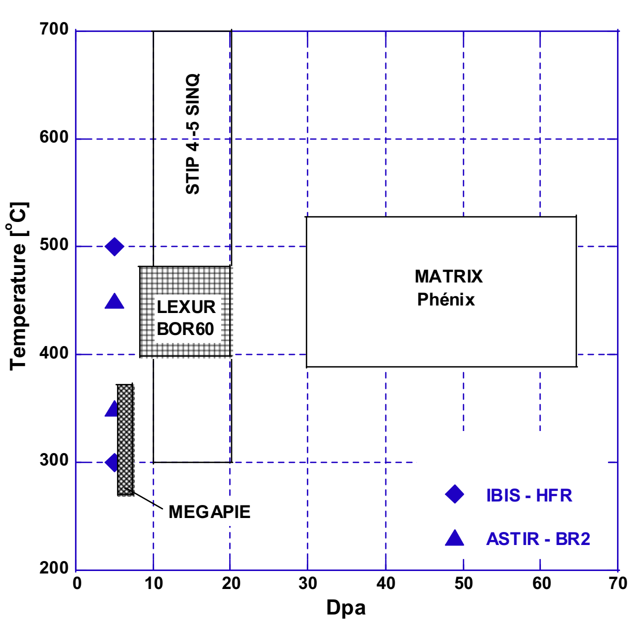

Fig. 1. Temperature and dose range (in dpa) of ongoing irradiation experiments in Europe relevant to the GETMAT project.

More recently, the T91 steel was irradiated to relatively high dose (42 dpa) with fast neutrons in the BOR 60 reactor and the PIE performed [16]. The focus was on the effects of irradiation at low temperature (325 °C), where significant hardening and loss of ductility was confirmed to occur in martensitic steels [16,33]. In order to investigate the higher irradiation temperature range of prime interest for GEN IV in-core structures, the MATRIX experiment was built [9] and the irradiation is currently ongoing in the Phénix reactor. A maximum irradiation dose of 60–70 dpa is planned, with irradiation temperatures ranging from 390 to 530 °C. In addition to the reference T91 steel, other conventional 9Cr steels, such as T92 (with improved high temperature creep properties), 9Cr-Eurofer, reference steel for Fusion Technology (for comparison), and GESA aluminised T91 and T92 steels, were also included in this experiment.

In the case of ADS, in addition to the core components, the spallation target is also a major concern. The internal structures (window, funnel, collector, etc.) will undergo a high rate of atomic displacements (dpa) induced by the spallation neutrons, as well as the production of some chemical elements via transmutation (i.e. helium, hydrogen), that are expected to contribute to the degradation of materials properties. Some data on irradiation effects in this environment in F/M steels were obtained in the past decade, partly in the US (irradiation at LANSCE) and mostly in Europe with the STIPs (SINQ Target Irradiations Programmes) conducted in the Swiss spallation target at PSI [34]. However the specimens were irradiated at relatively low irradiation temperatures (up to 350 °C). Therefore, new irradiation experiments are ongoing in order to irradiate samples at higher temperatures (up to 500–550 °C for the F/M specimens) to a maximum dose of 20 dpa and 1800 appm He.

For the ODS Fe–Cr alloys, the irradiation data on alloys developed for high temperature (non nuclear) applications are rather scarce and some of the published results are for materials which were not optimised in terms of composition/fabrication route. In the recent past, most of the work was performed on the reduced activation 9Cr-Eurofer ODS alloy developed within the Fusion community, as well as on the ODS MA957 alloy. High dose data are available for this last material and for an irradiation temperature of 325 °C (BOR 60 reactor experiment) [16]. In addition to ODS Eurofer, other ODS materials considered to be amongst the best Fe-based ODS alloys produced so far, such as the PM2000 alloy (material containing 20 Cr), produced by PLANSEE, as well as the MA957 (14% Cr) and MA956 (20% Cr) ODS materials, produced by INCO, are included in the MATRIX experiment. These same ODS alloys are also irradiated in the SINQ spallation target at temperatures up to 700 °C and a maximum exposure of 20 dpa/1800 appm He. The PIE of the ODS specimens irradiated both in Phénix and SINQ are part of the GETMAT PIE programme.

3.3.2. Synergistic irradiation and heavy liquid metal coolant effects

In addition to irradiation induced modifications of the dimensional and mechanical properties, the synergistic effects of irradiation and corrosion/embrittlement due to the coolant is a major concern, in particular for systems cooled by heavy liquid metals as e.g. ADS and LFR. Few experiments have been reported to address this issue, one notable exception being the LISOR experiments, in which T91 samples in flowing Lead Bismuth Eutectic (LBE) were subjected to tensile stress and irradiated with 72 MeV protons. However, the maximum total dose received by the LISOR samples was less than 1 dpa [34]. On the other hand, following the irradiation phase of the MEGAPIE experiment [8], the planned PIE of the target will provide unique data regarding the combined effects of irradiation in a proton–neutron spallation environment, corrosion/erosion/embrittlement by flowing LBE and cyclic thermal/mechanical loading on the properties of T91 steel.

Specimens are also being irradiated in a neutron spectrum and in contact with static LBE in the BR2 and HFR reactors for exposure up to 5 dpa at temperatures ranging from 300 to 500 °C [9]. However, data to higher dose and in a fast neutron spectrum are needed for the design of future LFR and to support the design of the subcritical blanket of the ADS. Therefore, a new irradiation of T91 steel is proposed in the BOR 60 reactor, to be performed in liquid lead with a maximum exposure of 20 dpa.

Table 4 summarises the PIE programme foreseen in the GETMAT project, which includes F/M steels, in particular T91 and T92 including coated specimens, as well as the above-mentioned ODS steels irradiated in the MATRIX and STIPs experiments. A large data collection will result from these activities, and will be a valuable complement to the experiments already performed in previous programmes and outlined above. In particular, for the reference T91 steel, the irradiation induced modifications of mechanical properties including tensile, impact, toughness and creep behaviour will be obtained for a large irradiation temperature and dose range, in both fast neutron and spallation environments.

Furthermore, the PIE of MEGAPIE, ASTIR/IBIS and LEXURII will provide a wealth of unique data on the combined effects of heavy liquid metals and irradiation on the performance of F/M steels, where the presently available knowledge is extremely scarce.

3.4. Multi-scale modelling and experiments

The planned modelling effort in the project aims at understanding the physical mechanisms occurring under irradiation of mainly FeCr alloys, as model alloys for high-Cr F/M steels. For this purpose, a multi-scale modelling approach, including computer simulation tools and the extensive use of advanced experimental techniques, is used. In recent years atomic-level modelling has provided keys to the interpretation of both early and recent, often puzzling, experimental observations concerning the behaviour of FeCr alloys under irradiation (see e.g. [35]), such as the non-monotonic trend of radiation hardening and embrittlement [36], or swelling [37], in FeCr alloys vs. Cr content, where a minimum is found in the 3–10 wt% Cr range.

Comprehension is the essential prerequisite for the development of reliable models beyond empirical correlations and for this reason atomic-level modelling in these alloys has been pursued, particularly in the framework of fusion research, in Europe, US and Japan. At the same time, a number of projects in Europe have addressed the problem of the development and experimental validation of multi-scale models for materials used in fission nuclear power plants, for example Zr alloys [38], low-alloyed reactor pressure vessel (RPV) steels and austenitic stainless steels [39].

These projects, particularly PERFECT [39], have strongly contributed to boost the collaboration on radiation damage modelling in Europe and have produced a number of outstanding scientific results, bringing important contributions not only to the understanding of the physical processes taking place in materials under irradiation, especially Fe alloys, but also to how they can be modelled. PERFECT has brought a wealth of data, both from theory and experiment, for the parameterisation of models of use for diluted ferritic alloys. A large number of ab initio calculated characteristic energies concerning point-defects and their clusters in Fe, as well as their interaction with a number of solute elements, including carbon, have been produced. Advanced interatomic potentials, based on new fitting methodologies, have been developed for Fe and a number of binary alloys [40,41]. The role of displacement cascades on the microstructure evolution vis-à-vis coarse-grained models based on Monte Carlo or rate theory techniques has been clarified. Atomic-level Monte Carlo models capable of treating a number of solutes and both vacancies and self-interstitials have been developed and applied. The foundation of dislocation dynamics models for Fe has been set. From the experimental point of view, one of the most complete microstructural characterisation on model alloys for RPV steels has been performed, and it revealed a number of mechanisms and effects that models should include and reproduce. PERFECT has also provided a much clearer vision of the potential and limitations of the existing modelling tools, thereby setting solid bases on which to build future projects, aiming at applying the multi-scale modelling approach for similar applications. In addition, within both SIRENA and PERFECT (projects fully devoted to modelling), attention has been focused on the aspect of the integration of models onto a common platform, revealing all the difficulties related to this task.

There are thus solid bases to address the development of quantitative models describing the microstructure evolution under irradiation and thermal ageing in concentrated FeCr alloys with a certain expectation of reliability, having in mind the specific situations for Generation IV or ADS. The development of the models for these alloys and the translation of the current level of qualitative understanding into quantitative predictions will demand an enormous amount of work in model development and parameterisation, which is expected to represent a large part of the activities.

Table 4
PIE programme underway or planned for the GETMAT project.

Experiment/reactor	MATRIX Phenix	STIP 4–5 SINQ	MEGAPIE	ASTIR BR2	LEXUR II BOR60	IBIS & SUMO HFR
Spectra	Fast neutrons	High energy protons and neutrons	High energy protons and neutrons	Thermal and fast neutrons	Fast neutrons	Thermal and fast neutrons
Materials	T91, T92, EUROFER T91,T92 GESA treated ODS (9–20 Cr)	9–12 Cr welds ODS (9–20 Cr)	T91 AISI 316L	T91, T91 GESA treated, welds, ODS	T91, T91 treated and welds, SS316L	T91, Eur-ODS T91 coated, SS316L, welds
Tests examinations	Tensile Impact CT, fractog. SEM, TEM	Tensile bending Charpy, SPT TEM	Tensile, bending, SPT, SIMS, XPS, XRD SEM, TEM	Pressurised tubes, CT, tensile, Charpy	Tensile corrosion	Tensile KLST SEM, TEM
Irradiation temperature	390–530 °C	300–700 °C	250–375 °C (T91 beam window)	350, 450 °C	400 °C: 316L 480 and 550 °C: T91	300, 500 °C
Dose range	30–65 dpa	10–20 dpa	~7 dpa (T91 beam window)	~ 5 dpa	Up to 20 dpa	2 dpa
Environment	Na	Inert gas	Pb–Bi	Pb–Bi, inert gas	Pb, inert gas	Pb–Bi Na (SUMO)

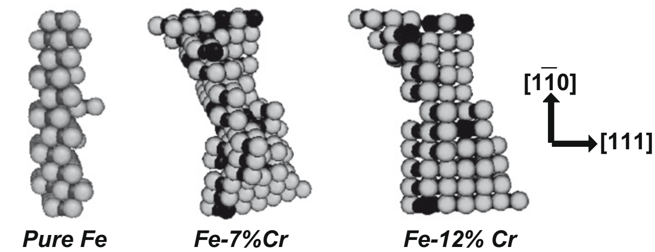

Fig. 2. Pictorial representation of a cluster of 91 self-interstitials in Fe and FeCr alloys. Light gray balls are Fe atoms, black are Cr and dark gray small ones are the lattice site shared by two atoms (one of the interstitial) [D. Terentyev and L. Malerba, SCK CEN Report, BLG-1026, June 2006].

This problem is particularly delicate because the materials considered here are concentrated solid solutions. Developing satisfactory microstructure evolution (Monte Carlo and rate theory) and radiation hardening (dislocation dynamics) models for concentrated alloys represents the main challenge of the modelling WP of the GETMAT project, as this is a field that has been hitherto little explored. The presence of a high concentration of solute atoms, which interact strongly with radiation induced defects (e.g. a self-interstitial loop) transforms any material property because of the local environment changes. A pictorial example of this is provided in Fig. 2, where a cluster of 91 self-interstitials is shown as it appears at the atomic-level in Fe and FeCr alloys. In Fe such a defect is a platelet of parallel crowdions (here represented as two 'light gray' atoms sharing the 'dark gray' lattice site). In FeCr the attraction between Cr and crowdion distorts the defect into a three-dimensional feature. Its formation and migration energy will depend strongly on how the Cr atoms are distributed around it. Accounting for this local environment dependence, especially in coarse-grained models where atoms are not explicitly treated, is the issue of main concern within this work-package. Since approximations will be needed, and effective average values will have to be used, model experiments become important for model development and validation. These are meant not merely to be a reference for model benchmarking, but also and more importantly to provide data, and therefore further understanding, about the effect of different important variables (temperature, dose, Cr concentration, etc.), thereby guiding the modelling priorities and in some cases helping in their parameterisation. The correct, targeted choice of these experiments and their careful performance is another fundamental task, to be performed in such a way as to be linked and complementary to similar activities already underway, e.g. within the fusion programme.

4. Summary and perspectives

The GETMAT project started in February 2008 and will last for five years. The consortium is composed of 24 partners from 11 European countries. Further important elements of the project are training activities, where workshops and thematic schools will be organized. Also, a user group will be formed with the aim of exchanging information about the evolution of the system designs and contribute to the design selections.

GETMAT will have an important impact on the improvement of knowledge on ODS alloy fabrication, shaping and joining/welding and their performance in a neutron irradiation field, where data are very limited at present. In addition, since the fabrication of ODS materials is a critical issue and little industrial experience is available in Europe, the consortium has identified, after a careful evaluation of the few data available, two different fabrication routes and two different ODS alloy compositions to be developed and tested within the project. The test matrix has been set in such a way that the outcome will help assessing these alloys under extreme thermal, mechanical and chemical conditions and will pave the way for the future of ODS alloy development for nuclear application in Europe. This step will be performed with the help of the user group, which has the role of evaluating the results produced by the consortium, iterating with design teams and exploiting these results within practical applications.

The 9–12Cr F/M steels experimental activities aimed at the qualification of this class of material are underway within national and international projects. The test – matrices proposed within this project aim at the extension of the available database. In particular, the highest effort has been put on ongoing irradiation experiments (e.g. MATRIX, ASTIR, IBIS and MEGAPIE) where the material is subjected to neutron irradiation fields in contact with liquid metals. In this way the project will have a greater impact with an optimised resource investment.

Special attention has been put on the qualification and ranking of promising welding technologies, which are essential for the construction of the nuclear power plants, and on corrosion protection methods, e.g. 'smart coatings', in order to address specifically the corrosion of the structural materials, thereby reducing the burden on the structural materials.

The modelling effort included in the project will provide bases for physical understanding and explanation of the experimental results, very much needed since the operating conditions of most future reactor concepts cannot be simulated in any existing facility, and extrapolation exercises will eventually be needed. Large dependence on modelling corresponds to the strategy of most major laboratories worldwide, aimed at minimising the time and resources for the development and qualification of structural materials for future nuclear reactor concepts. The project will therefore have an important impact on this methodology.

Finally, GETMAT is expected to contribute significantly to the material projects in preparation within the different system agreements of Generation IV. It is in particular expected that many of the activities of the present project will constitute the EU contribution to those collaborative Gen-IV projects. To ensure this contribution, specific links to international initiatives has been defined.

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