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between national programmes on advanced nuclear materials**

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EERA JPNM R&D Roadmap

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Summary

This Strategic Research Agenda (SRA) has been prepared by the EERA-JPNM based on a wide consultation with the scientific and also industrial community involved, to identify the research lines to be pursued in the EU to ensure that suitable structural and fuel materials are available for the design, licensing, construction and safe long-term operation of GenIV nuclear systems. Emphasis has been put on the fast neutron spectrum systems considered in ESNII, namely SFR, HLM-cooled systems (ADS and LFR), and GFR. However, links with other GenIV systems, namely (V)HTR, SCWR and MSR have been clearly established and effort has been devoted to identify GenIV materials research issues that are also of common interest for other nuclear and also non-nuclear energy technologies. This was done with a view to optimising the use of available resources, whenever possible, by joining forces with other research communities. Importantly, the content of this SRA is fully consistent with other relevant strategic documents and roadmaps compiled by other platforms and in other frameworks.

Grand challenges have been identified and, based on the current status of the research in the field of nuclear material science and on the needs of the industrial stake-holders, this document defines the specific scientific and technical objectives, the categories of materials to be studied, the research approaches and the timeframe to address them. The document treats also necessary corollaries to the proposed research activities, namely: infrastructures needed, need for education and training and mobility schemes, industry and regulators involvement, importance of international cooperation, estimate of resources required.

The materials considered here cover as priority the needs of the ESNII prototypes and demonstrators, but attention is put also on materials solutions that are intended for FOAK and commercial GenIV systems, in which higher energy efficiency and longer burnups than in the prototypes are targeted.

The research activities are organized in blocks that result from the application, for structural and fuel materials, of a well-established materials science approach, based on the combination of three classes of activities: (1) materials testing and characterization for full qualification and definition of design rules in a pre-normative spirit; (2) development of mechanistic and physical models in support of materials behaviour correlations used in design rules and improvement of materials properties; (3) development of advanced materials through experimental screening of solutions, also assisted by models rooted in the understanding of the physical processes that govern materials behaviour.

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List of abbreviations

| | |
|---------|---|
| ADS | Accelerator-Driven System |
| AFA | Alumina Forming Austenitic (steels) |
| ATF | Accident Tolerant Fuel |
| APT | Atom Probe Tomography |
| BU | Burn-up |
| CSE | Creep-Strength Enhanced |
| CSP | Concentrated Solar Power |
| CTR | Carbothermal reduction |
| CVD | Chemical Vapour Deposition |
| CVI | Chemical Vapour Infiltration |
| DBTT | Ductile-brittle transition temperature |
| DoW | Description of Work |
| dpa | Displacements per atom |
| DS | Deployment Strategy |
| EC | European Commission |
| EELS | Electron Energy Loss Spectroscopy |
| EERA | European Energy Research Alliance |
| EII | European Industrial Initiative |
| ENEN | European Nuclear Education Network |
| EOL | End Of Life |
| ESNII | European Sustainable Nuclear Industrial Initiative |
| ETIP | European Technology and Innovation Platform |
| ETP | European Technology Platform |
| E&T(&M) | Education and Training (and Mobility) |
| EU | European Union |
| FCCI | Fuel-Clad Chemical Interaction |
| FCMI | Fuel-Clad Mechanical Interaction |
| FeCrAl | Alumina forming F/M steels (from the main composing elements) |
| FEM | Finite Element Models |
| FIB | Focused Ion Beam |
| FGR | Fission Gas Release |
| FOAK | First of a kind |
| FPC | Fuel Performance Codes |
| F/M | Ferritic/martensitic |
| GenIV | Generation IV |
| GFR | Gas-cooled Fast Reactor |
| HCP | High Performance Computing |
| HEA | High Entropy Alloys |
| HIP | High Isostatic Pressing |
| HLM | Heavy Liquid Metal |
| HRTEM | High Resolution Transmission Electron Microscopy |
| HT | High Temperature |
| HTR | High Temperature Reactor |
| ICME | Integrated Computational Materials Engineering |
| ICT | Information and Communication Technology |

| | |
|----------|--|
| IEA | International Energy Agency |
| IR | Integrated Roadmap (SET-plan) |
| JPNM | Joint Programme on Nuclear Materials |
| LBE | Lead-bismuth Eutectic |
| LFR | Lead-cooled Fast Reactor |
| LME | Liquid Metal Embrittlement |
| LT | Low Temperature |
| LTO | Long-Term Operation |
| LWR | Light Water Reactors |
| MA | Minor Actinides |
| MAX | (M- transition metal, A- A group element, X- C or N) |
| MOX | Mixed uranium-plutonium OXide |
| MoU | Memorandum of Understanding |
| MR | Energy Materials Roadmap (SET-plan) |
| MS | Member State |
| MTR | Material Testing Reactor |
| M2F | Expert Group on Multiscale Modelling of Fuels of the WPMM of the OECD/NEA |
| NDE | Non-Destructive Examination |
| NEA | Nuclear Energy Agency |
| NPP | Nuclear Power Plants |
| NUGENIA | NUclear GENII/III Association |
| NC2I | Nuclear Cogeneration Industrial Initiative |
| OCJ | Oxide-Clad Joint |
| OCR | Oxide-Clad Reaction |
| ODS | Oxide Dispersion Strengthened (or Strengthening) |
| OECD | Organisation for Economic Co-operation and Development |
| PAS | Positron Annihilation Spectroscopy |
| PIE | Post Irradiation Examination |
| PLD | Pulsed Laser Deposition |
| PyC | Pyrolytic Carbon |
| SANS | Small-Angle Neutron Scattering |
| SCW | SuperCritical Water |
| SCWR | SuperCritical Water Reactor |
| SEM | Scanning Electron Microscopy |
| SETIS | SET-plan Information System |
| SET-plan | Strategic Energy Technology plan |
| SFR | Sodium-cooled Fast Reactor |
| SL | Surface Layer |
| SNETP | Sustainable Nuclear Energy Technology Platform |
| SPS | Spark plasma sintering |
| SRA | Strategic Research Agenda |
| SRIA | Strategic Research and Innovation Agenda |
| TEM | Transmission Electron Microscopy |
| TMT | Thermomechanical treatment |
| TRL | Technology Readiness Level |
| (V)HTR | (Very) High Temperature Reactor |
| WPMM | Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems of the OECD/NEA |
| XRD | X Ray Diffraction |

2DS

2 Degree Scenario

Executive Summary

With a yearly production in excess of 850 TWh_e, nuclear energy is the single largest source of low-carbon energy in the EU. Consistently, in the 2^o IEA scenario the worldwide nuclear energy production should increase from 11% in 2015 to 16%/15% in 2050/2060. Thus, **nuclear energy plays an important societal role, together with renewables, to ensure the energy transition**. Yet, three main open issues remain: (1) sustainable use of resources; (2) accident risk; (3) long-lived nuclear waste.

The sustainability of nuclear energy will be ensured by deploying GenIV systems. These can (i) produce more fuel than they consume, guaranteeing low-carbon energy production for millennia; (ii) offer much higher thermal efficiency than current reactors; and (iii) provide higher standards of passive safety, resulting also economically attractive. However, materials will be exposed to high levels of temperature and irradiation, in contact with potentially aggressive non-aqueous coolants, and the design needs to be extended to 60 years of lifetime. Thus, **the development, screening and qualification of suitably performing and affordable structural materials are crucial to make GenIV reactors an industrial and commercial reality, with positive impact on safety, waste, economy and sustainability of nuclear energy.**

This **Strategic Research Agenda (SRA)** has been prepared by the EERA-JPNM based on a **wide consultation with the scientific and also industrial community** involved, to identify the **research lines** to be pursued in the EU to ensure that **suitable structural and fuel materials are available for the design, licensing, construction and safe long-term operation of GenIV nuclear systems**. Emphasis has been put on the fast neutron spectrum systems considered in ESNII, namely SFR, HLM-cooled systems (ADS and LFR), and GFR. However, links with other GenIV systems, namely (V)HTR, SCWR and MSR have been clearly established and effort has been devoted to identify GenIV materials research issues that are also of common interest for other nuclear and also non-nuclear energy technologies. This was done with a view to optimising the use of available resources, whenever possible, by joining forces with other research communities. Importantly, the content of **this SRA is fully consistent with other relevant strategic documents and roadmaps compiled by other platforms and in other frameworks.**

Grand challenges have been identified and, based on the current status of the research in the field of nuclear material science and on the needs of the industrial stake-holders, this document defines the **specific scientific and technical objectives**, the **categories of materials to be studied**, the **research approaches and the timeframe to address them**. The document treats also necessary corollaries to the proposed research activities, namely: infrastructures needed, need for education and training and mobility schemes, industry and regulators involvement, importance of international cooperation, estimate of resources required.

A close link to the industrial application is essential. In particular the goals and the needs of the reactor designers must be very clear in order to support the processes of licensing and construction of advanced nuclear systems, for which return of experience is limited. The involvement of TSOs and regulators, to follow the procedures used for materials qualification and possibly guide them, is also desirable and likely to further accelerate the licensability of nuclear components. The connection with industry is essential in the process of development of new materials, as well, especially if innovative fabrication routes need to be explored that should, eventually, be upscaled to industrial production.

The materials considered here cover as priority the needs of the ESNII prototypes and demonstrators, but attention is put also on materials solutions that are intended for FOAK and commercial GenIV systems, in which higher energy efficiency and longer burnups than in the prototypes are targeted.

The research activities are organized in blocks that result from the application, for structural and fuel materials, of a **well-established materials science approach**, based on the combination of three classes of activities: (1) materials testing and characterization for full qualification and definition of design rules in a pre-normative spirit; (2) development of mechanistic and physical models in support of materials behaviour correlations used in design rules and improvement of materials properties; (3) development of advanced materials through experimental screening of solutions, also assisted by models rooted in the understanding of the physical processes that govern materials behaviour.

Besides the obvious need of adequate financial resources in order to address the research problems outlined in this SRA, a few **recommendations** emerge that this document intends to bring to the attention of stake-holders, particularly research managers and decision-makers:

R1: Data from materials property measurements after exposure are the essential ingredient for robust design curves and rules. Plenty of data were produced in the past that are *de facto* unusable either because covered by confidentiality or because not properly stored. Correct data management to guarantee their availability for future re-assessment is therefore essential and should be encouraged

and fostered. In particular, **financially supported policies to foster data sharing and to encourage old data disclosure should be implemented.**

R2: Some infrastructures are absolutely essential to enable the correct qualification of nuclear materials, not only irradiation facilities, but also suitable hot cells where materials can be safely handled and tested, nuclearized characterization techniques, loops and pools for compatibility experiments, etc. They are also crucial for education and training of young researchers and operators. These infrastructures are costly to build and maintain. Other research facilities are, on the other hand, more common and sometimes redundant. **A rational and harmonised, trans-national management of infrastructures in Europe, including schemes for facility sharing, would be highly desirable and, at the end of the day, beneficial for all.**

R3: International cooperation with non-EU countries where research on nuclear materials is pursued can be very valuable for Europe. Quite clearly, the goals of this cooperation are in the end the same as in the case of internal European cooperation, namely coordinate activities, share data, have access to infrastructures. Currently, however, **the instruments available in Europe for international cooperation are not sufficiently attractive to motivate researchers, so more efforts should be made to improve their attractiveness and ease of application.**

R4: The **nuclear materials research community in Europe is currently strongly integrated** and engaged in thriving collaboration, in a bottom-up sense, much more than at the level of instruments offered to make this integration efficient and functional in a top-down sense. This **SRA** is largely the **result of matching bottom-up research proposals with top-down strategies.** The appropriate instrument to allow this community to deliver according to the goals should provide the conditions to implement the agreed upon research agenda and to set up suitable E&T&M schemes allowing knowledge, data, and facility sharing. Since the financial support of Euratom will never be sufficient, earmarked funding from the MS dedicated to support integrated research on nuclear materials is crucial. In this sense, **co-fund instruments, like a European Joint Programme, seem to be the most suitable ones.**

These recommendations are clearly based on the willingness of pursuing a policy of increased integration rather than of isolation, at all levels: research organisations, EU Member States, and European Commission. Besides the amount of resources that can be reasonably allocated, to cover a need that has been estimated to range -depending on the ambition of the goals- between 15 and 50 M€/yr, this requires finding a difficult equilibrium between the need to make the best use possible of the limited resources available, in a framework, nuclear energy, where support is politically not simple to obtain, and the legitimate ambition to preserve everyone's assets, in a context of healthy competition.

1. Introduction

1.1 Societal, economic and technical challenges

With a yearly production in excess of 850 TWh_e, nuclear energy is the single largest source of low-carbon energy in the European Union.¹ This corresponds to more than ¼ of the electricity in Europe, thereby guaranteeing secure and reliable base-load supply. Consistently, in the 2 degree scenario (2DS) of the International Energy Agency (IEA) the worldwide nuclear energy production should increase from 11% in 2015 to 16%/15% in 2050/2060.² Thus, nuclear energy plays an important societal role, together with renewables, in ensuring the energy transition. However, three main open issues remain: (1) sustainable use of resources; (2) accident risk; (3) long-lived nuclear waste.

Sustainability is not sufficient in current nuclear systems: only less than 1% of the fuel energy content is used in present day nuclear power plants (NPPs), more than 90% of which are light water reactors (LWR). Sustainability can be greatly enhanced by the deployment of fourth generation (GenIV)³ fast neutron reactors, along with the facilities that are needed to close the fuel cycle. The combination of fast reactor and close fuel cycle allows indeed the energy extracted from the available uranium resources to be maximised. Fuels irradiated in fast neutrons generate as much Pu from the ²³⁸U by neutron capture as is consumed by fission. The reactor cores can be thus optimized, by pushing the burnup to high values, i.e. by letting the fuel remain for longer in the reactor, to produce more Pu than they consume (breeder reactor). This can be then extracted and reused for refuelling.

Increasing the thermal efficiency is another key factor to improve sustainability in terms of use of resources. The thermal efficiency is the ratio between electricity and heat produced and its increase means not only a larger amount of electricity produced for a given thermal power, but also less waste heat and less environmental impact and need for cooling. Thermal efficiency depends on the temperature of the reactor core and on the performance of the conversion system. Present LWRs have thermal efficiencies under 33%, while modern coal plants reach approximately 39% and combined-cycle gas plants even 50 to 60%. The use of non-aqueous coolants, mandatory in fast neutron reactors, will allow operation at temperatures well-above those of LWRs, thereby pushing the efficiency close to coal plants and, in the case of very high temperature, combined-cycle. With liquid metals this can be achieved close to atmospheric pressure, with a design largely based on passive safety systems. This provides the designers an important tool to make fast breeder reactors both safe and economically attractive.

Fast reactors offer also an additional virtue, they have the ability to transmute the minor actinides (the elements representing a long term source of radiotoxicity and heat) into short lived fission products. In this way, again provided that high burnups are reached, the radiotoxicity of the waste can be reduced to time scales below 1000 years, which easily fulfil man made engineered repository licensing. Furthermore, the capacity of these repositories can be increased by a factor of 10 or more. In addition, the adoption of a closed fuel cycle requires only a short-term storage of the irradiated fuels before their reprocessing and reuse.

Thus, GenIV systems composed by fast reactors and close fuel cycle facilities create more usable fuel than they burn and can operate at high temperature using passive safety systems, thereby increasing enormously the efficiency in the use of resources and guaranteeing safe energy production for several centuries, with significantly reduced waste production.

In order to reach the above goals, however, materials in GenIV systems will be exposed to higher temperatures and higher irradiation levels than in today's LWRs. Moreover, the compatibility of materials with non-aqueous coolants needs in several cases to be demonstrated. The operating conditions for the fuel pins will be further complicated due not only to very high temperatures but also to massive temperature gradients, leading to major deterioration of physical properties, especially when combined with irradiation damage and presence of fission products. At the same time, the overall cost of these systems must be at par with other low-carbon energy systems, including current LWRs. Since the capital cost of the construction represent the largest part

¹ In 2014, 876 TWh_e were produced by nuclear. This, combined with 406 TWh_e of hydroelectricity covering especially peaks, corresponds to about 40% of the total electricity and represents by far the largest portion of low-carbon electricity in Europe. Wind comes third, with 253 TWh_e (source: Eurostat - http://ec.europa.eu/eurostat/statistics-explained/index.php/Electricity_and_heat_statistics). However, wind keeps growing and produced, in 2016, almost 300 TWh_e (source: Windeurope - <https://windeurope.org/about-wind/statistics/european/wind-in-power-2016/>).

² Source: Energy Technology Perspectives, IEA, 2016/2017.

³ Technology Roadmap of the Generation IV International Forum: <https://www.gen-4.org/gif/upload/docs/application/pdf/2014-03/gif-tru2014.pdf>

of the investment in the case of NPPs, to be economically viable GenIV reactors are expected to be designed and licensed for a 60 year lifetime. This is obviously quite a challenge, given in particular the harsh conditions that both structural and fuel materials are going to be exposed to. Thus, **the performance of nuclear materials is an essential point to make GenIV reactors a reality**. Furthermore, the development of materials with superior resistance to high temperature and exposure to aggressive coolants can also be beneficial for other energy technologies.

1.2 The EERA JPNM in the European platform landscape

The Joint Programme on Nuclear Materials, JPNM (www.eera-jpnm.eu), was launched in 2010 and is one of the 17 joint programmes (JPs) of the European Energy Research Alliance, EERA (www.eera-set.eu). Altogether, these JPs cover all low-carbon energy technologies and systems. EERA, created in 2008, supports the European Strategic Energy Technology (SET) Plan of the European Commission (EC), which was launched in 2007. It does so by coordinating the work of, in 2017, almost 180 public research organisations, towards the development and deployment of cost-effective low carbon technologies. The goal is to meet the sustainability targets set by Europe for 2020 and 2050, to counteract climate change and guarantee security of energy supply and competitiveness. One important defining feature of EERA is the focus on relatively low technology readiness levels (TRL⁴<5). EERA is indeed dealing mainly with research towards innovation, while the industrial implementation (TRL>5) characterizes the technology platforms and the industrial initiatives.

The European Sustainable Nuclear Energy Industrial Initiative (ESNII⁵), which has the task of developing GenIV fast neutron reactors in Europe using different technologies, was launched under the umbrella of the sustainable nuclear energy technology platform (SNETP⁶) at the same time as the EERA JPNM. **The EERA JPNM provides the R&D for the structural and fuel materials needed for the development and implementation of fast reactors in Europe**, as defined by ESNII. Currently, this is the main reason of existence of the JPNM in EERA. Figure 1 illustrates the connections between energy platforms, in particular nuclear energy platforms, in Europe.

However, the scope and goals of the JPNM go also beyond. By operating mainly at TRL<5, the EERA JPNM deals mainly with fundamental research, although projected towards specific technological applications and to bridging with the industrial initiatives via, mainly, pre-normative research. The SNETP recognises the importance of basic technology developments, because, as stated in its Deployment Strategy (DS) of 2015⁷ they “*open routes for the identification of common trunks for Gen II, III, IV and cogeneration application, notably in areas such as:*

Material behaviour for structural components and fuel

Structural integrity of systems and components

Manufacturing & assembly technology”

Thus, **basic research on structural and fuel materials’ behaviour belongs to one of the areas where commonalities through nuclear reactor generations and types can be actually found**. For this reason the SNETP explicitly mentions⁷ that “*the interface with EERA/JPNM should be reinforced for the development of new and innovative materials*”. **The memorandum of understanding (MoU) signed in December 2016 between EERA JPNM and SNETP concretises this intention, extending the collaboration** to enhance synergy not only with ESNII, but **also with the other SNETP pillars**, namely the Nuclear GENII/III Association (NUGENIA)⁸ and the Nuclear Cogeneration Industrial Initiative (NC2I)⁹. Moreover, several issues faced by materials for fission reactors are common to fusion reactors and systems, as well: hence the possibility of finding cross-cutting topics with this other, longer-term form of nuclear energy.

⁴ https://ec.europa.eu/research/participants/data/ref/h2020/wp/2014_2015/annexes/h2020-wp1415-annex-g-trl_en.pdf

⁵ <http://www.snetp.eu/esnii/>

⁶ <http://www.snetp.eu/>

⁷ <http://www.snetp.eu/wp-content/uploads/2016/01/SNETP-DEPLOYMENT-STRATEGY-2015-WEB.pdf>

⁸ <http://www.nugenia.org/>

⁹ <http://www.snetp.eu/nc2i/>

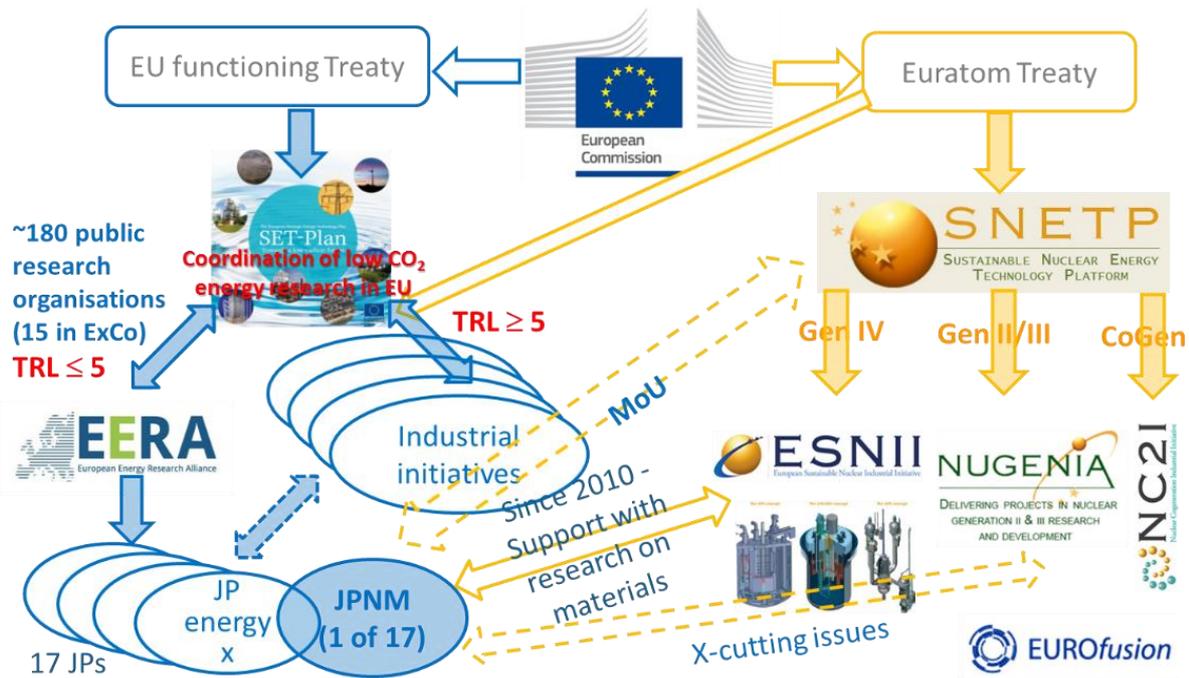


Figure 1: Illustration of the connection between (nuclear) energy platforms in Europe.

It is also clear that materials with superior properties in terms of high temperature and corrosion resistance may find their way to other energy technologies. In particular, **within EERA the JPNM finds natural grounds for collaboration with other joint programmes**, targeting high temperature operation in environmentally harsh environments.

1.3 Generation IV systems in Europe

In the vision of the SNETP the European nuclear industry can continue to deliver safe low-carbon nuclear energy for the present and the coming centuries, with a commitment towards even higher safety standards and sustainability, in two phases:

Safe extended operation of existing GenII/III nuclear power plants or long-term operation (LTO), including new builds;

Parallel gradual deployment of GenIV fission reactors and systems, guaranteeing more sustainable and safe nuclear energy, with the potential for nuclear heat generation.¹⁰

GenIV systems may be commercially deployed around the middle of this century. Four **GenIV fast reactor prototypes and demonstrators** with different maturity levels, as well as high temperature reactors (HTR) for cogeneration demonstration, are being studied in Europe.

The **sodium-cooled fast reactors (SFR) is the most mature technology, lead-cooled fast reactors (LFR) are considered the next technology** and present advantages in terms of passive safety and potential modularity, while **gas-cooled fast reactors (GFR) are a somewhat longer term alternative** that opens the way to even higher temperature and therefore efficiency. In ESNII, the SFR prototype is ASTRID, while ALFRED and ALLEGRO are prototypes for, respectively, LFR and GFR. In addition, MYRRHA is a flexible research facility for material testing and demonstration of accelerator-driven systems (ADS) for waste minimization, whose features are strongly related to LFR technology. ASTRID and MYRRHA are considered the front-runners in terms of time to construction. ALFRED and then ALLEGRO should follow later (see Table 1, where the ESNII roadmap in terms of GenIV prototypes and demonstrators and relevant fuel facilities is summarised, as in the SNETP 2015 deployment strategy⁷). Start of construction milestones are currently expected as follows, provided that appropriate financing is secured:

ASTRID and MYRRHA 2025-2030

ALFRED: 2030-2035

¹⁰ In the long run, the gradual insertion of fusion systems in the energy production market, in cohabitation with fission systems, may also occur.

ALLEGRO: beyond 2035

In addition to fast reactors, GenIV includes also thermal reactor concepts which aim at specific targets, namely: the supercritical water reactor (SCWR) as an advanced upgrade of existing LWRs; the already mentioned (very) high temperature reactor ((V)HTR) aimed at industrial heat production and cogeneration; and the molten salt reactor (MSR) as an especially proliferation-resistant system. These systems are not included in the ESNII portfolio, yet work on them is going on in Europe, mainly at member state level. Currently the SCWR is being included in the NUGENIA portfolio as a type of advanced LWR, while the (V)HTR is the reference for NC2I and a demonstrator is envisaged beyond 2025. The MSR is being more and more often mentioned as a very long term option that offers intrinsic advantages in terms of fuel cycle. These systems, while not in the focus of JPNM, may be addressed in terms of materials as part of cross-cutting activities, if a sufficient critical mass is created, mainly based on national programmes.

Importantly, **GenIV reactors should not be decoupled from the relevant fuel cycle facilities**, enabling the fabrication of MOX fuel and, in the longer run, the fabrication and use of fuel that contains minor actinides, including advanced fuels (e.g. nitrides), so as to reduce to the very minimum the waste, by burning it in the reactor, as well as in ADS. These facilities include obviously also the reprocessing and recycling, in order to guarantee sustainability for centuries to come. For these reasons, one should talk of GenIV *systems*, and not simply GenIV (fast) reactors.

Table 1: Summary of ESNII Roadmap concerning GenIV prototypes and demonstrators and relevant fuel facilities (from SNETP DS 2015).

| | T0 + 10 y | T0 + 20 y | T0 + 30 y |
|---|--|---|--|
| ASTRID | Basic design – license and start construction | Commissioning and operations – integration of feedback experience | Basic design , license and start construction of FOAK SFR |
| MYRRHA | Basic design – license and start construction | Commissioning and integration of feedback experience from operations | |
| ALFRED | Conceptual design – start basic design and licensing | Complete basic design – construction and commissioning | Basic design , license and start construction of FOAK LFR |
| ALLEGRO | Viability of GFR concept | Conceptual- basic design and licensing | Start construction and commissioning |
| Fast reactor MOX fuel cycle facility | Basic design- license and start construction of FR MOX fabrication | Conceptual design – licensing of a reprocessing/ recycling facility | Start construction and commissioning of advanced recycling facility |
| | | | Extend capacity of FR MOX fuel fabrication for FOAK FR |
| Transmutation | Fabrication of one Am bearing segment of fuel pin per year | Conceptual – basic design and licensing of a pilot plant of capacity one full Am (or MA) fuel assembly per year | Start construction and commissioning of pilot plant for Am / MA fuel fabrication |

1.4 Objectives and Grand Challenges of the JPNM

The objective of the EERA JPNM is to improve safety and sustainability of nuclear energy by focusing on materials aspects. This has two implications:

1. **Better knowledge of materials behaviour** under operating conditions, seeking predictive capability, to select the most suited materials and define for them safe design rules, especially considering radiation and temperature effects, while caring for compatibility with coolants.
2. **Development of innovative materials** with superior capabilities, either through suitable processing methods applied to existing materials or adoption of new types of materials, in terms of resistance to high temperature, irradiation and aggressive environments.

The EERA JPNM has published a vision paper, in which its view on nuclear energy materials is put forward in connection with the ESNII systems and three Grand Challenges are identified. These are based on the three pillars of the EERA JPNM's research approach and strategy (see Figure 2):

- Assessment of candidate structural and fuel materials and components in operational conditions: screening, selection and qualification, development of design rules;
- Development of advanced models to rationalise materials behaviour, underpin design rules and provide basis for the improvement of materials properties by providing predictive capability;
- Development of innovative structural and fuel materials and advanced fabrication processes for industrial use, with superior capabilities in terms of resistance to irradiation, high-temperatures and aggressive environment.

The **three grand challenges** correspondingly identified are:

Grand Challenge 1: Elaboration of design rules, assessment and test procedures for the expected operating conditions and the structural and fuel materials envisaged. This involves deployment of infrastructures for exposure to ageing, testing of materials and production of data and knowledge.

Grand Challenge 2: Development of physical models coupled to advanced microstructural characterization to achieve high-level understanding and predictive capability: an essential asset, given the scarcity of experimental data and the difficulty and cost of obtaining them.

Grand Challenge 3: Development of innovative structural and fuel materials and fabrication processes to achieve superior thermo-mechanical properties, better compatibility with coolants and improved radiation-resistance, in partnership with industry.

Addressing these Grand Challenges requires a concerted action at European level involving research community and industrial partners.

1.5 Timeframe, purpose and structure of this Strategic Research Agenda (SRA)

The SNETP Deployment Strategy (DS) of 2015 provides the timeline that can be currently given in the best case scenario for the different nuclear technologies. Two important messages can be deduced:

Research on materials that impact structural integrity, component ageing and advanced solutions for components in nuclear systems, **is a continuous process**: it does not have a deadline and constitutes the research humus on which innovation, and therefore in this case better safety and efficiency for nuclear systems, can grow;

In order to allow the **licensing and construction** of GenIV prototype systems, however, the research on materials needs to provide sufficient data for qualification and possibly codification of design rules in a horizon that, depending on the prototype and the specific issue, has a span of **no more than 5 to 15 years**.

This horizon, taking into account the specificities of nuclear energy, is extremely short and, to be met, requires the deployment, already now, of significant resources devoted to materials research and development, if the goal of sustainability has to be reached.

This SRA covers, mainly, the period 2016-2025, but it is in fact projected also beyond. Its main purpose is to identify and detail the scientific and technical gaps that need to be addressed in this timeframe, both in a short-term technological application perspective (prototypes, ~2025-2035) and in preparation of the longer-term technology (commercial deployment, beyond 2035).

The research programme of the EERA JPNM is defined by three documents, as shown in Figure 3.

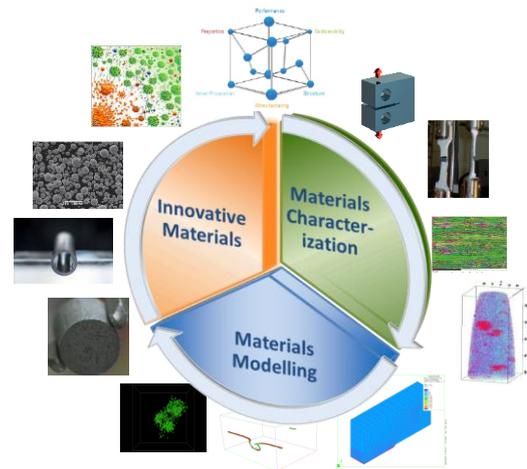


Figure 2: The research approach and strategy pillars of the EERA JPNM



Figure 3: Documents describing the research programme of the EERA JPNM with increasing level of detail from top to bottom.

The **vision paper** provides the general high-level context and is meant for any audience;

The **strategic research agenda (SRA)**, to be revised only periodically, targets both researchers and research funders and describes the research and development route to be followed (roadmap), according to the three strategy pillars of Figure 2, to face the three Grand Challenges of the EERA JPNM identified in the vision paper.

The **description of work (DoW)**, regularly revised, corresponds to the implementation of the SRA: it is embedded in the current situation and describes work that is actually ongoing, including tasks and deliverables at the level of subprogrammes, but extending also to the other strategic activities that accompany research.

Thus the present SRA is expected to be of use to guide the activities of all nuclear materials stakeholders: scientists, industries, research managers.¹¹ It is aimed at

researchers active within and also outside the JPNM, but it has the ambition to reach also other stakeholders, such as members of the SET-plan Working Groups, nuclear and non-nuclear technology platforms representing both industry and research organizations, as well as managers and decision-makers, especially member states representatives and European commission officers.

The roadmap that is part of the present SRA adopts a matrix structure, as proposed by C. Featherstone and E. O'Sullivan¹², schematically described in Figure 4, which is a visual representation of its “fabric”: the horizontal warp is given by the cross-cutting research strategies, or infratechnologies, while the materials addressed provide the vertical woof¹³.

Within each matrix element, or *block*, different issues and relevant goals are identified.

The three infratechnologies here identified are obviously interdependent. This is also the meaning of the virtuous circle of Figure 2, which exemplifies the JPNM research strategy. There is indeed a strong overlap in terms of methodology and needed infrastructures between pre-normative research on existing materials and screening of newly developed materials. It is also clear that modelling the physics that governs the behaviour of existing materials and the simulation of the features expected in the new materials supports and possibly accelerates both pre-normative research and development of new materials. This mutual feeding needs to be always kept in mind.

In this document, the research agenda is presented, including discussions of all the strategical aspects related with it, namely:

- actual roadmap in terms of description of key issues and relevant activities
- identification of cross-cutting research with other nuclear and non-nuclear energy technologies
- needs for specific facilities and infrastructures
- interaction with industry
- benefits of international cooperation
- some aspects related with education and training.

¹¹ http://ec.europa.eu/research/industrial_technologies/pdf/research-road-mapping-in-materials_en.pdf

¹² C. Featherstone and E. O'Sullivan, A review of international public sector strategies and roadmaps: a case study in advanced materials, Centre for Science Technology and Innovation, Institute for Manufacturing, University of Cambridge, March 2014.

¹³ “Research Road-Mapping in Materials”, A. F. de Baas (Ed.), European Commission, DG Research, EUR 24210 EN (2010), doi: 10.2777/87000.

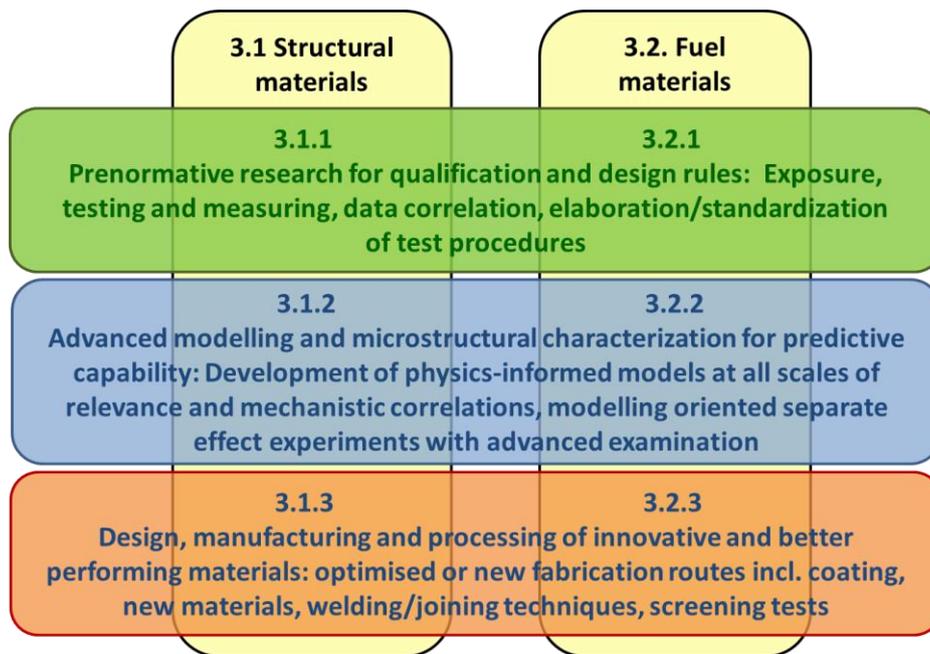


Figure 4: Matrix structure of the roadmap: the crossing of infratechnologies and materials identifies blocks which form the sections of the roadmap (see chapter 3).

1.6 Sources and links with other roadmaps and SRAs

To elaborate the present SRA, the priorities of ESNII concepts and the objectives of SNETP were taken into account, while identifying commonalities with other nuclear and also non-nuclear energy technologies. To this aim the EERA-JPNM launched, in 2015, a wide consultation involving ESNII representatives, to report on the most recent design of the prototypes, and materials scientists active in the research performed in the six sub-programmes of the JPNM. Subsequently, contacts were taken with SNETP pillars¹⁴ and the fusion community, as well as with other joint programmes in EERA¹⁵. A MoU was signed between EERA and SNETP in December 2016, the main purpose of which is to agree officially on common research topics. As first action for the implementation of this MoU, technical annexes that consensually identify common research activities between EERA JPNM and SNETP pillars have been produced. Moreover, other roadmaps, SRA and roadmapping initiatives were taken into account. These sources are listed here.

1.6.1 Materials roadmap enabling low carbon energy technology (MR)

This roadmap was issued after a wide consultation among stake-holders launched by the EC in December 2011.¹⁶ It highlighted the steps to be followed in the field of materials for advanced low-carbon energy technologies, defining a 10-years European R&I agenda. For nuclear fission it proposed an R&D programme on commercially available materials for the prototypes and demonstrators, and on advanced materials for industrial scale systems. This programme targeted materials for cladding, coatings for enhanced corrosion and erosion/wear resistance and novel advanced materials. Specifically: (i) manufacturing and out of pile testing of F/M 9%Cr steels for heat exchanger, (ii) manufacturing and out of pile (and possibly in-pile) testing of oxide dispersion strengthened (ODS) claddings, (iii) manufacturing and out of pile (and possibly in-pile) testing of SiC_f/SiC composites cladding and (iv) manufacturing and testing of coatings. **Several technical issues mentioned in the MR are also the focus of this SRA, which updates them to the current perception of needs.**

¹⁴ More specific joint task descriptions with SNETP pillars than in this SRA can be found in the Technical Annexes of the EERA JPNM/SNETP MoU.

¹⁵ <http://www.eera-set.eu/eera-inter-jp-cross-fertilization-workshop-on-materials-for-energy-applications-and-technologies-publication-of-the-report/>

¹⁶ https://setis.ec.europa.eu/system/files/Materials_Roadmap_EN.pdf

1.6.2 The SET-plan Integrated Roadmap (IR)

Nuclear energy is recognized by the SET-plan as a low-carbon energy source. Accordingly, in the Integrated Roadmap (IR), that was officially presented and launched on the occasion of the SET-plan conference in December 2014 in Rome, Heading 5 of Part II reads: “*Supporting Safe Operation of Nuclear Systems and Development of Sustainable Solutions for the Management of Radioactive Waste*”. Under this Heading, nuclear materials for GenIV reactors are explicitly mentioned: “*Qualify nuclear materials for operation under Gen IV conditions and develop innovative materials to improve plant safety and efficiency*”. Indirectly, moreover, research and innovation on nuclear materials are implicit in other parts of the document under the same Heading, where they are necessary for the accomplishment of the objectives mentioned. The present SRA provides essentially an expansion of the IR, to allow its practical implementation for what concerns research on GenIV nuclear materials.

Among the 10 key actions identified by the SET-plan on the basis of the IR and used as a starting point for discussions with Member States and stakeholders on the prioritisation of energy research activities in Europe in 2017, nuclear energy enters as *10th objective*, as follows: “*Maintaining a high level of safety of nuclear reactors and associated fuel cycles during operation and decommissioning, while improving their efficiency*”. **Materials have clearly an important role to play to reach this 10th SET-plan target and its relevant implementation plan.**

1.6.3 OECD/NEA Technology Roadmap for Nuclear Energy and other OECD/NEA initiatives

The OECD/NEA Technology Roadmap for Nuclear Energy 2015¹⁷ recommends that, in the timeframe 2015-2030, the “*governments [are] to recognise the long-term benefits of developing GenIV systems [...]*”, this , as mentioned, involving crucially the selection and qualification of materials. The present SRA is thus a timely document, to ensure efficient use of the funding to be provided by governments.

At the time of preparation of this SRA, the OECD/NEA is engaged in the elaboration of a Nuclear Innovation Initiative (in the low-carbon perspective) with a horizon to 2050, which is due to be issued by the end of 2017 / beginning of 2018. **Via the MoU signed between EERA JPNM and OECD/NEA a strong link was created between the present SRA and the priorities and projects identified in this NEA initiative.**

1.6.4 SRIA and DS of the SNETP

The SNETP released a Strategic Research and Innovation Agenda (SRIA)¹⁸ in February 2013 and the corresponding DS in June 2015. Together, these documents define clearly the three pillars for nuclear energy research and demonstration in the following decades, as follows:

Support the fully safe operation of present and newly built LWR, so-called GenII/III reactors, allowing the development of sustainable solutions for the management of radioactive wastes;

Prepare the development and demonstration of advanced fast neutron GenIV reactor technologies associated with a closed fuel cycle to enhance the sustainability of nuclear energy;

Promote the use of nuclear energy beyond electricity generation, namely in cogeneration of heat or hydrogen production or water desalination.

The importance of transversality between SNETP pillars and with other energy platforms is one of the main messages of the SNETP DS. In particular “*the identification of R&D project clusters for Gen II, III, IV and cogeneration applications for basic technology developments, e.g. performance and ageing of NPPs for long-term operation and high reliability components for structure and fuel, could valuably build a bridge between the different nuclear system developments and with other ETPs as well*”. In other words, **materials behaviour for structural components and fuel and more generally the structural integrity of systems and components are common trunks through GenII/III and IV, and cogeneration as well**. Consistently, in the present SRA, significant effort is made to identify cross-cutting issues between materials for GenIV and for GenII/III, as well as for high temperature. The EERA JPNM operates at low TRL (<5), while industrial initiatives, involving utilities, necessarily deal especially with higher TRL solutions and approaches: this is a criterion to identify overlaps and set boundaries, as done in the Technical Annexes to the MoU between SNETP and EERA JPNM.

¹⁷ <https://www.oecd-nea.org/pub/techroadmap/>

¹⁸ http://www.snetp.eu/wp-content/uploads/2014/05/sria2013_web.pdf

1.6.5 Other roadmaps

Other documents connected with the present SRA in terms of issues, materials and goals, consulted during its preparation, are: (i) the SRA of EuMaT, the European Technology Platform for Advanced Engineering Materials and Technologies¹⁹; (ii) the roadmap of Materials Modelling of the European Materials Modelling Council²⁰; the Roadmap of the Metallurgy Europe – EUREKA cluster²¹.

¹⁹ <http://eumat.eu/filehandler.ashx?file=11580>

²⁰ https://emmc.info/wp-content/uploads/2015/04/EMMC_Roadmap_V3.0.2.pdf

²¹ <http://metallurgy-europe.eu/>

2. Nuclear materials for GEN IV systems and research approaches

2.1 Material degradation processes and relevant properties

As stressed in the previous section, GenIV concepts will involve very harsh conditions in terms of temperature, irradiation levels and interaction with the coolant for all materials. Here we focus on structural and fuel materials.

Coolant operation **temperatures** between 400°C and 600°C in the case of liquid metal-cooled systems, and targeting 850°C in gas-cooled systems, which will lead to temperatures around 2200°C at the centre of the fuel in normal conditions and exceeding 1000°C in structural materials in off-normal conditions, coupled to tremendous **temperature gradients**, inflict severe **thermomechanical stresses** on the fuel and plant components. These **trigger the degradation of both structural and functional materials, through several simultaneously occurring mechanisms**. Moreover, **cooling fluids** represent invariably a **chemically hostile environment**, the exposure to which also affects severely the performance of materials in contact with them. Finally, nuclear materials are exposed to varying levels of irradiation, the number of displacements per atoms (dpa) reaching 1 in the fuel in less than 1 day in reactor, exceeding 100 in the cladding during its stay in the reactor, though being less than 2 in the vessel during the whole reactor life. These levels of **irradiation invariably affect negatively the performance of the materials**.

The **mechanical degradation mechanisms** that are known to operate at high temperature are creep, thermomechanical fatigue, creep-fatigue, ratcheting and thermal ageing. Thermochemical mechanisms due to interaction between fuel, structural materials and cooling fluids are corrosion (oxidation), dissolution, erosion. In some cases exposure to fluids may affect also the mechanical properties (e.g. liquid metal embrittlement). Irradiation affects the materials by changing substantially and continuously their chemical composition, microstructure and microchemistry. Importantly, these mechanisms act simultaneously, thereby introducing a high complexity in terms of synergic effects that cannot be simply “linearly superposed”.

For **fuel**, the main parameter that needs to be maintained against these mechanisms is the temperature at the centre of the material, which needs to be less than its melting temperature, itself strongly affected by the modifications undergone by the fuel in reactor.

For materials with **structural function** the properties that need to be preserved are: (i) strength and resistance to creep deformation, with rupture beyond design lifetime and stress; (ii) toughness and resistance to crack initiation and propagation also under cyclic loading (fatigue) (iii) especially for cladding materials, high thermal conductivity and limited thermal expansion, as well as hermeticity against gaseous and volatile fission products. In addition, materials need to maintain thickness against the effect of the coolant environment..

Only a few classes of materials can potentially meet these requirements, with different levels of quality in their response: these are described in the next section.

2.2 Nuclear materials and components addressed

2.2.1 **Structural materials**

Structural materials are those used to fabricate a component that bears load or stress, whichever its origin (mechanical, thermal, vibrations...).

In nuclear reactors it is important to distinguish between two types of structural components, replaceable and non-replaceable:

- Replaceable components are *designed to be easily extracted from the reactor or even their replacement is an essential part of the reactor operation*. The most obvious examples of this type are ,the fuel elements which are periodically reshuffled or removed, when the burnup allowed by neutronics and materials has been reached. In general these elements are the hollow tubes containing the fuel pellets, the bundles holding them together and the structures that support them. A significant deviation from this general description of the fuel element may be found in HTR or GFR, where the high temperatures advice the use of ceramic materials.

The **replaceability generally goes hands in hands with more severe degradation and therefore shorter lifetime**. This is the case of the **fuel cladding**, which is exposed to the highest irradiation dose, experiences the highest temperatures and temperature gradients and is in contact with the coolant and the fuel.

- Non-replaceable components constitute the main structure of the reactor. Major examples are the vessel or, in pool reactors, the upper cover of the containment. These components are characterized by the fact that their replacement, though theoretically not impossible, essentially corresponds to building a new unit, i.e. it is so costly and complex that it is economically not affordable.²² These components need to be designed for the full lifetime of the reactor or, the other way round, their lifetime defines the lifetime of the reactor itself.

Non-replaceable components need to be designed in such a way that the degrading agents are mitigated as much as possible, so that the materials ageing is slow enough to guarantee that the component remains fit for its purpose until the end of the life of the plant. Typically, for example, the vessel will be at a sufficient distance from the fuel elements to be subject only to marginal or no irradiation and the system will be designed so that the temperature will be lower than close to the fuel element. In addition, it is sometimes possible to apply on the vessel protections from the effect of the coolant, although the problem of prolonged exposure to the coolant may in fact be the main lifetime limiting factor for irreplaceable components.

The choice of the most suitable materials for a given component and application comes obviously to a large extent from previous experience. **The current design of GenIV prototypes and demonstrators planned in Europe envisages the use of austenitic steels as main class of structural materials**, specifically 316L(N) for most components, including the vessel, and 15-15 Ti for the cladding and other fuel element parts. Depending on the specific prototype/demonstrator also other materials may enter, e.g. ferritic/martensitic (F/M) steels in the SFR core. In particular for the GFR, ceramic materials or perhaps refractory metallic alloys must be considered for the core. Other materials of foreseen use for out-of-core components, such as the intermediate heat exchanger, the steam generator, turbine blades, coaxial pipes or hot gas headers, are Ni-based alloys. However, austenitic steels are clearly the current dominant choice across system prototypes. The reason is that they are a very good compromise between several requirements, even without excelling specifically in any of them. But the crucial reason for this choice is the **return of experience** from their use in fast reactors built and operated in the past, like Phénix and Superphénix in France. There exists, therefore, a wealth of experimental data on them, based on which **design rules have been already established and introduced in standard codes**. This is a significant advantage towards design and licensing. Nonetheless, still several aspects need to be qualified in austenitic steels and require intensive research effort, especially concerning compatibility with coolants. **Other materials**, on the other hand, **need full qualification and codification before being considered for design**.

F/M steels (e.g. T91, EM10, HT9, reduced activation F/M like Eurofer or F82H for fusion, ...) **offer a number of desirable properties as cladding and core materials, superior to those of austenitic steels**, namely better thermal conductivity, lower expansion coefficient and better resistance to radiation-induced void swelling. The third point is key: while austenitic steel cladding will not sustain more than 100 dpa,²³ F/M steels are expected to reach 200 dpa without experiencing any swelling. **However, they can currently operate only below 550°C, suffer from radiation-induced embrittlement below 350-400°C and may become brittle in contact with heavy liquid metals (HLM)**²⁴. Reliable correlations and models for the identification of design rules accounting for all these phenomena are needed to use F/M steels in GenIV systems. Otherwise, the resistance to high temperature and embrittlement of F/M steels need to be improved.

Three main paths are pursued to improve the properties of F/M steels: (1) production of **oxide-dispersion strengthened (ODS) steels**, currently using powder metallurgy techniques; (2) tuning of the composition together with appropriate thermo-mechanical treatments (TMT), in a conventional metallurgy framework (**creep-strength enhanced, CSE**); (3) **surface protection** from hostile coolants. The use of ODS F/M cladding, explicitly foreseen e.g. in the second phase core of ASTRID, would allow the upper limit of the temperature window to be increased beyond 700°C while providing effective improvement in terms of radiation resistance. There are, however, still several technical difficulties to be overcome, that concern mainly the ODS steel fabrication process, in particular the lack of an established industrial production. The second way to improve the properties of F/M steels, via thermo-mechanical treatments (TMT) and composition tuning, is very attractive because the manufacturing process remains within conventional metallurgy. This approach, based on the idea of engineering stable microstructures, has been used successfully outside the nuclear field, leading to 650°C as upper limit of the operating temperature range of steels that, however, are not nuclear grade. Finally, composition tuning with Al addition is a way to improve compatibility with coolant, via formation of a surface layer of alumina (alumina forming alloys, both ferritic –FeCrAl- and austenitic – AFA-). FeCrAl alloys

²² In the history of nuclear energy, however, some components initially meant to be irreplaceable have been eventually replaced, because the high cost was counterbalanced by the expected gain from the exploitation of the plant.

²³ Currently, 15-15 Ti is supposed to reach 90 dpa, better versions of it might allow this limit to increase up to 110 dpa.

²⁴ Liquid lead and its alloys, such as Lead-Bismuth Eutectic (LBE) or Pb-Li, are collectively denoted as heavy liquid metals (HLM).

can be strengthened by ODS, giving rise to a potentially very promising material, while AFA steels would inherently exhibit good creep properties through NiAl precipitation. In alternative, only the surface composition may be changed with the addition of aluminium, or ceramic coatings can be applied on the metallic substrate.

For **temperatures above 800°C**, such as those **targeted in GFR and HTR**, no steel is able to maintain its fitness for purpose, possibly with the exception of high Ni austenitic steels such as alloy 800. In this case, **Ni-based alloys** (e.g. Inconel 617, Haynes 230 and Hastelloy XR) **are a possibility for components outside the core**, particularly heat exchangers, not only for GFR and HTR. However, these alloys suffer from severe irradiation embrittlement and also swelling, so their use in the core can be critical. Other **refractory materials** need therefore to be considered and the spectrum is quite wide, ranging **from refractory metals (e.g. molybdenum or vanadium) to ceramics**. The latter offer in general very attractive properties in terms both of stability to high temperature and resistance against wear and corrosion/erosion. However, both are **penalized by brittleness**. Thus, refractory materials for structural components are likely to be mainly composites, enabled by fibres or other reinforcements to exhibit some type of pseudo-ductile mechanical behaviour. The **most intensively studied ceramic material is SiC_f/SiC**, i.e. silicon carbide fibres in silicon carbide matrix. SiC_f/SiC is the main candidate material for GFR cladding, it is a material of use for the HTR and it is a suitable candidate for LFR and SFR cladding, as well. **Perspective ceramic or metallic refractory materials**, including innovative ones, that can be considered for cladding or other applications in GFR/HTR, but also for other systems, are: ODS-Mo, high entropy alloys (HEA), and MAX phases. For all these materials, including SiC_f/SiC, there are issues concerning fabrication processes, optimal choice of composition/architecture of the component, joining, standardization of testing, and the usual problems of radiation resistance in contact with coolants.

Based on this *excursus* on structural materials for GenIV systems, Table 2 attempts a classification and a prediction of the main structural materials that are expected to be used in the different GenIV systems currently part of ESNII, distinguishing different time phases, from prototype/demonstrator (phase I and II) to FOAK and finally commercially deployed plants. **This table is important because it allows the different materials and therefore the research and development devoted to them to be set in the correct timeframe and in connection with a specific system**. Of course the actual timing for each system is not the same (see Table 1) and will crucially depend on the priorities politically associated with one system or another and the readiness with which funding is allocated: none of this can be easily predicted. However, from a conceptual viewpoint the phases schematised in the table remain valid and therefore, through this table, it is possible to establish priorities and perspectives.

Table 2 is not expected to be fully comprehensive of all materials foreseen for the prototypes: it simply intends to highlight on which materials or classes or materials the JPNM research is focused and where these materials fit, in order to immediately understand the application and context. This table, based on current knowledge and perception about the future, will require regular updates. Phases I/II of the prototypes are not necessarily defined in the ESNII plans and can be of very different nature. In ASTRID and probably MYRRHA the second phase will have continuity with the first one, except switching to different cladding materials (tested and qualified during the first phase), to improve the performance and/or the reliability of the system. The GFR distinguishes two (or more) phases, as the first demonstrator will not rely on the full qualification of the materials needed for high temperature operation. In the LFR there is currently no established division in two phases, but core materials evolution will be unavoidable to improve performance and reliability. Finally, concerning FOAK and commercially deployed reactors, the materials mentioned in the table are largely a speculation. The message is that some materials will not be used for the prototypes, but remain a targets of the EERA JPNM because of their long term potential.

Table 2: Main structural materials expected to be used in the different ESNII GenIV systems, distinguishing different phases that define the timeline, from prototype/demonstrator to FOAK and finally commercially deployed plants.

| PHASES → SYSTEMS ↓ | | ESNII Prototype | | FOAK | Commercial deployment |
|------------------------|----------------------------------|--|--|--|---|
| | | As licensed (phase I) | Evolving (phase II) | | |
| ASTRID (SFR) | Periodically Replaced Components | AuSS: 15-15Ti – AIM1 (cladding) F/M: EM10 (wrapper) | AIM2, F/M ODS (cladding) | TMT F/M, F/M ODS | TMT F/M, F/M ODS, perhaps SiC/SiC |
| | Permanent Structural Components | AuSS: AISI316L(N); 800SPH | | AuSS: AISI316L(N); TMT F/M | |
| MYRRHA (ADS) | Periodically Replaced Components | AuSS: 15-15Ti – 1.4790 (cladding) | Coated 15-15Ti (FeAl, FeCrSi, FeTa, ...), AFA, perhaps MAX (as coating?) | N/A | |
| | Permanent Structural Components | 316L(N) | | | |
| ALFRED (LFR) | Periodically Replaced Components | Al ₂ O ₃ PLD coated 15-15Ti (cladding, FA structures) | Coated 15-15Ti (Al ₂ O ₃ PLD, FeCrAl, FeCrSi, Max phases), perhaps AFA | Most likely AFA, perhaps FeCrAl ODS | AFA, FeCrAl ODS, or perhaps coated Mo-ODS, or SiC/SiC , |
| | Permanent Structural Components | 316L(N) Ferritic steel lined with AFA AISI316L (ASTM) / 15-15Ti + Al diffusion coating | | AFA, ODS-FeCrAl, Ferritic steel lined with AFA? | |
| ALLEGRO (GFR) / (V)HTR | Periodically Replaced Components | GFR: T<550°C: 15-15Ti (cladding) – AFA?/ HTR: TRISO (SiC) | GFR: T> 850°C: SiC/SiC (cladding) / HTR TRISO (SiC) | SiC/SiC, perhaps Mo-ODS/ HTR TRISO (SiC) | |
| | Permanent Structural Components | T<550°C: 316L(N) – AFA ?/ HTR: graphite | GFR: 550<T<850°C: AFA, FeCrAl / HTR: graphite | GFR: AFA or FeCrAl, perhaps , Mo or V alloys / HTR: graphite | |

2.2.2 Fuel materials

Nuclear fuel is a consumable and removable component at the heart of the reactor. It **is the seat of the fissions of nuclei, which produce the energy** to be ultimately used to produce electricity. It remains several years in the reactor until the maximum burn-up admissible has been reached. It is **exposed to the harshest irradiation and temperature conditions**. Together with the cladding, however, it is the only component whose performance can be significantly improved during the lifetime of a reactor.

The fuel materials are combined with cladding in the fuel elements. By conception, nuclear fuels and fuel elements must in particular

- Provide the power expected during their whole stay in reactor.

- Use the fissile elements as best as possible to reduce the cost of energy production
- Confine the fission products inside the fuel elements in all operating and accidental conditions

Nuclear fuels and fuel elements differ widely from reactor to reactor, in geometrical configuration, fuel composition and cladding. **Fissile atoms used in nuclear reactors, however, are mainly uranium and plutonium.** Fuel materials are thus compounds of these elements, either refractory ceramics such as oxides, carbides, nitrides or silicides, or stable metallic alloys. More complex systems include ceramic/ceramic or ceramic/metal composites, or molten salt fuels. As to geometries, fuel rods, fuel plates and pellets have been developed. Actinide oxides (U and mixed U-Pu oxides, MOX) are the most industrially used fuel materials and have been used in power reactors since the 1960s.

MOX will be the fuel for the first cores of ALL ESNII fast reactor prototypes. Variations of this fuel were used in previous European fast reactor programmes: Phenix, Superphenix, and Dounreay fast reactors. MOX fuel is Europe's only major knowledge and competence base for fast reactors, even if all European fast reactors are now closed. It is thus natural that this fuel is the first choice for the future.

MOX fuel crystallises in the cubic fluorite structure, so isotropic behaviour can be expected. It has high melting point, though only moderate thermal conductivity, resulting in **typical operating temperatures at 80% of the melting point.** Its manufacturing technology is known and proven, although it needs new validation when a new plant is deployed.

Though widely studied in the past, the MOX product can bear important intrinsic hallmarks linked to the fabrication methods used, e.g. porosity distribution, grain size, impurity levels, all of which can come to bear in its performance. This is not surprising when one considers the combination of the material alteration in the neutron flux, and the massive temperature gradients (up to 500°C/mm). Furthermore, reactor core designs have evolved, with particular effort based on thermal hydraulics and neutron physics to reduce such safety relevant parameters as void coefficients, so that pellet geometry has evolved from the classical pellet, to thicker pellets with an annulus to limit centre line temperatures. Though knowledge is available from the past, it is not usable directly, and must be leveraged to enable the licensing of future MOX fuels for first ESNII cores. Furthermore, the ESNII first cores, in addition to their differing reactor coolants, will not always be operated under the same power rating as in the past, necessitating further reactor specific investigations.

The introduction of multi-recycling of Pu will also perturb the Pu concentration in the fuel, but more importantly higher concentrations of ²⁴¹Pu in the Pu isotopic vector will lead to higher contents of ²⁴¹Am, should significant delays between separation and insertion in the reactor occur. This will lead to increased demands on the fuel fabrication technology (remote and shielded handling) and on the in pile performance as helium will be produced in greater quantities than for today's typical values.

In the longer term, the reduction of the long term toxicity of the waste can be dramatically improved by the introduction of advanced nuclear fuel cycles within which the minor actinides (MA) – americium, neptunium and curium – are extracted from the spent fuel and introduced in the fuel cycle for their transmutation in fast reactors.

Two types of concepts are envisaged for transmutation.

The first concept, known as **homogeneous mode**, involves diluting minor actinides in standard fast-reactor fuel. The advantage is the strong similarity between the structures of the various actinide oxides (fluorite-type cubic structure) and their mutual solubility. To minimize the impact of the introduction of minor actinides on reactor safety parameters, fuel minor actinide content is kept relatively low (a few % of heavy atoms). In such conditions, only a slight evolution is anticipated in fuel behaviour and performance, the properties remaining quite close to those of standard fuel, which facilitates the qualification of these fuels.

In the second concept, the **heterogeneous mode**, the minor actinides are concentrated in specific assemblies (also called minor actinide bearing blankets) located at the periphery of the reactor core. This results in a limited perturbation of the core behaviour. In addition, the MA bearing assemblies are manufactured in dedicated plants, separately from standard fuel, which enables limited quantities of MA-bearing fuels to be handled. The low neutron flux level experienced at the periphery of the core, however, slows down the transmutation process and this is compensated by increasing the MA fraction up to 15-20%. A large R&D effort is required for the design of these specific objects and to ensure their qualification.

Because of the high neutron emission, thermal power and toxicity of the minor actinides, the fabrication of MA-bearing fuels requires heavy shielding. A down selection of optimal fabrication routes must be achieved and remote handling for fuel fabrication and assembly production engineered and qualified. Then, the assessment of the irradiation performance of these fuel types, performed in the frame of several European projects since the 90's, needs to be completed.

In a further long term step, optimising core performance in terms of breeding and increased margins to melt would necessitate the adoption of mixed uranium and plutonium carbides and nitrides (denoted MX = MC and MN). These fuels have less moderation, and lead to harder neutron spectra and shorter doubling times. In addition, they have high melting points and significantly higher thermal conductivity than MOX fuels, so that they operate at about 40-50% of their melting point, which provides an appealing safety aspect. Their fabrication is not trivial, if high purities are to be achieved, and there are question marks about their volatility at temperatures below the melting point, a matter increasing in significance for Pu multi recycling, as the built in Am component could be (but not fully proven) even more volatile than the U and Pu constituents.

Similarly to Table 2 for structural materials, Table 3 lists the fuels that are envisaged to be used in the different ESNII systems in the different time phases. The same caveats apply as for Table 2.

Table 3: Main fuel materials expected to be used in the different ESNII GenIV systems, distinguishing different phases that define the timeline, from prototype/demonstrator to FOAK and finally commercially deployed plants.

| SYSTEMS ↓ | PHASES → | | FOAK | Commercial deployment |
|--|----------------------------------|---|---|-----------------------|
| | ESNII Prototype | | | |
| | As licensed (phase I) | Evolving (phase II) | | |
| ASTRID (SFR) | MOX 20-25% Pu from UOx recycling | MOX 20-25% Pu from MOX recycling, MA bearing fuels? | MOX 20-25% Pu from MOX multirecycling, MA bearing oxide fuels | |
| MYRRHA (ADS) | MOX with up to 30-35% Pu | MOX, MA bearing fuels | N/A | |
| ALFRED (LFR) | MOX with up to 30% Pu | MOX, advanced MX | Advanced MX, MA bearing fuels | |
| ALLEGRO (GFR) / (V)HTR ALLEGRO (GFR) / (V)HTR | MOX | MOX, advanced MX | Advanced MX, MA bearing fuels | |

2.3 Materials research approaches

One of the main duties of materials engineers is to be able to foresee timely when, due to degradation, the material is no longer fit for its purpose, under any kind of external interference. This implies knowing what happens to the material and how its properties change while in operation, i.e. knowing the degrading agents and processes and their effects. This knowledge, which can be defined as **qualification of the material**, is generally obtained by **testing** the material or the component under relevant conditions, as well as by **monitoring** it during operation.

The acquired knowledge of the behaviour of a material is then crystallised in **models**, that can be more or less detailed, specific or general, empirical or theoretical, or a mixture of these. **Tests and monitoring** are necessary **to verify up to what extent models are followed** in each specific case. In turn, **models are needed to interpolate or extrapolate to conditions under which tests cannot be performed**. Since materials in operation can, in some cases and within limits, be monitored using non-destructive techniques, but cannot be continuously tested, models are necessary also to plan the maintenance and the inspection of the component and to foresee its lifetime.

The knowledge of the behaviour of materials, finally, leads naturally to the **identification of ways to improve their response under given conditions**, by either protecting them, or improving their properties through appropriate processing, or by identifying or even developing in a targeted way entirely new materials.

These three aspects of materials research define the approaches (*infratechnologies*) of interest for this SRA, i.e. methods and techniques that are common to structural and fuel materials and appear as the warp of the research agenda of Section 3, as illustrated in Figure 4 **Figure 4: Matrix structure of the roadmap: the crossing of infratechnologies and materials identifies blocks which form the sections of the roadmap (see chapter 3).** They are also linked and constitute a virtuous circle, as illustrated by Figure 2. They are described in detail in this section as applied to nuclear materials, in particular for GenIV systems.

2.3.1 Materials qualification: design rules and codes, fuel performance codes

As said, GenIV system prototypes and demonstrators, certainly in their first phase, will necessarily be constructed using structural and fuel materials that are already largely available today, on which some return of experience exists from previously operated reactors. However, innovative nuclear design such as for the **ESNII systems can rely on very limited return-of experience**, depending on the system.²⁵ Thus the **materials selected need to be qualified for the conditions expected in the prototypes and demonstrators, including off-normal conditions** that may be faced during accidents and to which they need to be proven to resist for sufficient time: high temperature, high irradiation levels, and compatibility with coolants, for instance between HLM and structural components.²⁶ In turn, the actual **operating conditions of prototypes and demonstrators will be influenced and determined by the properties of these materials**, after applying suitable safety margins, to avoid their degradation beyond acceptable limits.

Off-normal conditions are especially important in the nuclear field in the aftermath of the Fukushima accident, as a consequence of which stricter safety requirements are imposed and more effort is needed to demonstrate structural integrity in accident scenarios. Accident scenarios translate into load excursion with high temperatures and load rates. The temperature range for mechanical properties assessment needs therefore to be extended and strain rate properties in the relevant range are necessary. If such data do not exist, then specific research programmes are needed.

In this context **materials qualification** means *generation and maintenance of evidence to ensure that material or equipment will operate on demand, under specified service conditions*, meeting system performance requirements.²⁷

Materials qualification is the pre-requisite for the establishment of robust design rules for structural components and fuel performance codes and is therefore defined as **pre-normative research**, where *pre-normative* refers to research aimed at *establishing standards or common procedures to ensure that materials and components in nuclear reactors are designed and operated in accordance with the best available engineering practices and the current scientific knowledge*.

Materials qualification requires dedicated experiments both out of pile and in pile,²⁸ to collect comprehensive and reliable data of the material properties relevant for the operational conditions.

Even for out-of-pile experiments, **suitable infrastructures are needed** where, for example, candidate materials are exposed for increasing times to a wide range of temperatures, in contact or not with specific coolants, stagnant or flowing. After or during exposure, these materials need to be suitably tested and examined to verify the level of degradation they exhibit, in terms of changes of properties of engineering interest.

Test standards are developed for this purpose by organizations such as ASTM, CEN and ISO and prescribe how tests should be conducted and how the data should be analysed to assure consistent material properties, irrespective of where the tests are performed.

Since exposure times are necessarily limited, **accelerated tests are often a must**. In some cases of interest for GenIV systems, the characterisation may require the identification of new and bespoke standard procedures to execute exposure and, especially, tests. These procedures are especially delicate in the case of accelerated tests, because their relevance for real operating conditions needs to be proven.

²⁵ SFRs have been built and operated in the past and the Design Code RCC-MRx was specifically developed to support the development of these fast reactors. Thus ASTRID can to a large extent be based on the experience from the French SFR reactors Rapsodie, Phénix and Superphénix. For the lead- and gas-cooled reactors, however, there is very limited, if any, return of experience and the safety and other requirements must be demonstrated by pre-normative research.

²⁶ HLM compatibility is not included in any code today and it is quite urgent to develop codes and standards. The two key HLM related degradation mechanisms are corrosion and liquid-metal embrittlement.

²⁷ Since the requirements for nuclear applications are very strict and the operational conditions harsh, special nuclear grades of materials are often developed, AISI 316L(N) is an example.

²⁸ Pile is the core of a reactor, where the materials are exposed to neutron irradiation.

In-pile experiments imply in principle repeating the same type of exposures, but under **irradiation**, preferably in **fast neutron spectra** in the case of GenIV systems, up to the dose expected in service. In the absence of suitable facilities, **Material Testing Reactors (MTRs)** with predominantly thermal neutron spectrum can be used, with irradiation limited to lower dose, counting on the possibility of safely extrapolating to different spectra and higher doses.²⁹

Eventually, the data gathered for each candidate material through this expensive procedure need to be rationally translated into **robust design rules for components and laws and models for fuel performance codes**.

For **structural materials**, data from standard tests are processed from a design standpoint; for instance a lower envelope is obtained for scattered material data for fatigue curves or accelerated laboratory creep data are extrapolated to the operational conditions, corresponding to lower stresses and temperatures. Design rules eventually comprise closed-form equations and strict criteria, together with basic material properties analysed as described. They are developed to be conservative and simple to apply at the design stage. An important concept is "allowable stresses" that depend on the material and the temperature and should ensure that deformations are in the elastic regime. Eventually, **design rules are collected in design codes**, together with materials specifications & design data for different materials and components, by making reference to test standards and qualification procedures.

The RCC-MRx code has been developed specifically to support the SFR technology and has been identified as the most appropriate design code for all ESNII reactors. It consists of a single document that covers in a consistent manner the design and construction of components for high temperature and research reactors and the associated auxiliaries, examination and handling mechanisms and irradiation devices. The design rules were developed to cover the mechanical resistance of structures close to neutron sources that can, depending on the situation, also operate in significant thermal creep conditions. It is divided in three main sections:

- Section I contains general provisions common to the entire code;
- Section II gives additional requirements for the alternative use of rule sets applicable for non-nuclear classified components and special instructions for component subject to specific regulation;
- Section III: is a set of applicable rules organized in 6 Tomes:
 - Tome 1 contains design and construction rules and comprises alphanumerically numbered subsections. Subsection Z with 20 technical appendices contains for instance basic design material properties, welded joints, elasto-visco plastic analysis and defect assessment.
 - Tomes 2 to 5 contain the rules corresponding to various technical areas: procurement of metal products; destructive and non-destructive examination methods; welding requirements; other fabrication operations such as cutting, forming and surface treatments.
 - Tome 6 contains a collection of probationary phase rules which do not yet have sufficient feedback in the standard code.

RCC-MRx does not contain any specific rules for environmental effects except thinning, and material property curves and design rules are based on 40 years operational life. It is updated every three years. Other codes are the R-codes developed to support the plant life management of UK reactors, R5 for high-temperature problems and R6 primarily from defect assessment. There are some commonalities between the RCC-MRx appendices and the R-codes. It should be noted that Design Codes such as RCC-MRx and ASME BVPH only address structural materials and components. Hence fuel claddings are not included in these codes.

In the case of the **fuel element**, which includes the cladding, the main design tools are **fuel performance codes (FPC)**, which enable the simulation of the thermal and mechanical behaviour and evolution of the fuel element in reactor as a function of the irradiation and thermal parameters in normal operational, incidental and accidental conditions. To this aim, these codes solve coupled partial differential equations governing heat transfer, stresses and strains in the fuel element, the evolution of isotopes and the behaviour of various fission products in the fuel rod with boundary conditions defined by the reactor operational conditions. These equations involve material properties for the fuel and the cladding, which evolve with the residence time in reactor. Models are therefore needed to describe the very complex relationships between the evolution of these properties and the relevant parameters, especially the temperature, composition and microstructure.

Such modelling is essential to understand and interpret the measurements carried out in reactors and the results of post-irradiation examinations, to predict the behaviour of specific fuels and/or in specific operating conditions and to demonstrate the satisfactory behaviour of fuels in all operational situations to support safety reports.

²⁹ No neutron irradiation facility currently exists anywhere in the world that allows doses on the order of 100 dpa to be reached within reasonable times and at affordable costs. Such facilities did exist in the past, though.

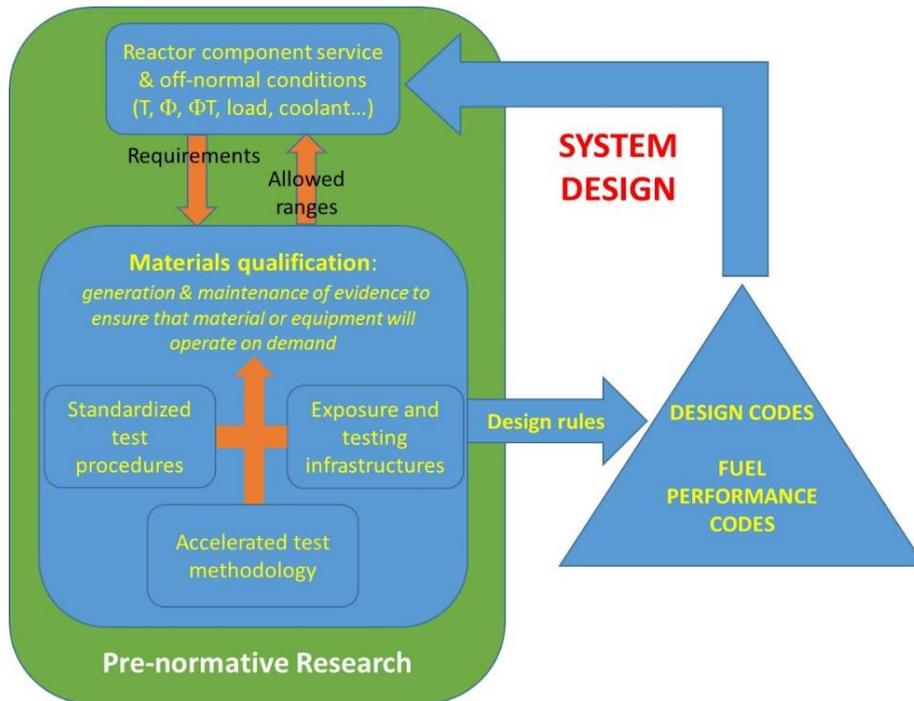


Figure 5: Schematic representation of the main elements that enter the process of generation of component design codes and fuel performance codes.

The development of FPC started quite empirically, relying on simple laws derived from experimental data. Progressively, more and more irradiation data were capitalized in the codes and more physics was introduced in the code. It is now universally recognized that a better understanding of the underlying phenomena of fuel behaviour is the prerequisite for a significant improvement of the codes.

The main European codes for GenIV reactors are TRANSURANUS developed by JRC in collaboration with various institutions through Europe, and Germinal co-developed by CEA and EDF.

Figure 5 schematically illustrates the process of materials qualification and generation of design rules and laws that should enter codes. The relevant infratechnology described here corresponds to the first line in the warp of Figure 4: “Exposure, testing and measuring for qualification and design rules”, concisely referred to as **materials qualification**.

2.3.2 Advanced materials modelling and characterisation

Exposing, testing and measuring to collect data of engineering relevance is not sufficient. The exposure to real conditions in laboratory is challenging in terms of infrastructures, time, costs, and also know-how and the data collected cannot cover all possible conditions, nor be fully representative of real environments. In particular, exposure times comparable with the lifetime of the reactor,³⁰ or high irradiation levels such as those reached by cladding and fuel, are hardly accessible in a laboratory, or the dose-rate will be much higher. Also, the combination of effects and their synergy is difficult to reproduce in a laboratory. So, the **extrapolation of data is unavoidable, but purely empirical extrapolations may be tricky or even unreliable** and therefore ultimately dangerous. Nevertheless, **in the past materials have been largely qualified** in this manner, describable **as a paradigm of “observe and qualify”**: the observation of the materials performance under a variety of conditions was the main ingredient in their qualification and licensing. This practice, though still used today, is **gradually undergoing a paradigm shift, whereby the materials are subjected to “design and control”**. This paradigm shift mitigates costs and reduces lead times to deployment.

At the heart of the design and control paradigm lies the greater reliance on advanced modelling and simulation, partly generated by improved theory, but also by the vastly increased computational power in the last decades, crucially coupled with advanced microstructural and micromechanical characterization, using ever more powerful techniques for materials examinations and testing at all

³⁰ 1/3 is considered sufficient for extrapolation, but it can also be a challenging duration/dose.

scales. This approach, lately denoted as *integrated computational materials engineering* (ICME), can provide robustness and predictive capability, initially underpinning, then gradually improving and finally replacing traditional empirical approaches, such as those used in current fuel performance codes or in dose-damage correlations for the vessel of existing LWRs.

In short, the ICME approach aims to **reach truly predictive capability by being able to describe in a fully physical way**, possibly with a single tool fed by more fundamental models, **the evolution in time of both the microstructure** (redistribution of lattice defects) **and the microchemistry** (redistribution of chemical species) of materials exposed to irradiation and/or high temperature (including exposure to coolants). **The output of these microscopic evolution models should then be the input of models that operate at the meso- and macroscopic length scales and predict accordingly the corresponding changes of materials behaviour and properties.**

As an example, this would be the ideal link of models to predict radiation effects at the microscopic scale:

- Models to predict damage production in collision cascades. Techniques of application here are electronic structure and interatomic potential calculations using molecular statics and dynamics.
- Atomistic models that allow mobility and stability to be studied for all radiation defects involved in the microstructural and microchemical evolution. Techniques of application are as above, but extend to atomistic kinetic Monte Carlo models.
- Parameters and mechanisms determined as above are the base of microstructural/microchemical models that describe how radiation damage evolves versus dose. Techniques of application here are object kinetic Monte Carlo, rate theory-based cluster dynamics, phase field, ...

All these models and especially those devoted to describe microstructural/microchemical evolution require suitable **data for validation and also calibration from modelling-oriented experiments**, i.e. experiments in which materials are exposed as for qualification, but are designed to understand mechanisms, by separating variables and effects, rather than to measure engineering properties. Here the **use of combined modern microstructural and microchemical characterization techniques**, such as transmission electron microscopy (TEM) in all of its multiple forms,³¹ atom probe tomography (APT), small angle neutron scattering (SANS), positron annihilation spectroscopy (PAS), and many others, **plays a crucial role.** Experiments of this type, in which key variables such as temperature, irradiation dose and dose-rate need to be accurately controlled and varied over sufficiently wide ranges, are invariably delicate, and may be long and costly. Like in the case of qualification, specific facilities are needed for irradiation and subsequent characterisation. In this context, **charged particle irradiation** (ions, protons, electrons ...) **is a very valuable and affordable tool.**

In order to bridge from the microscopic description to higher scales, especially those involving dislocations (involved in all plastic deformation processes, from tensile properties to creep) and then to continuum mechanics, the following approaches may be used:

- Calculation of the interaction of gliding and climbing dislocations with both pre-existing (e.g. grain boundaries) and radiation induced microstructural features (not only single point defects but also voids, precipitates, solute clusters and dislocation loops), as a function of type of feature, type of dislocation, temperature and strain rate. Techniques of relevance here are atomistic ones, especially molecular statics and dynamics with interatomic potentials, supported by electronic structure calculations.
- Transfer the above understanding in the form of local rules, to dislocation dynamics models, applicable at single-crystal level.
- Derive and/or parameterize, from dislocation dynamics and other tools, appropriate constitutive laws for crystal plasticity models, applicable to aggregates of crystals (grains), that should contain and reproduce the effects of the microstructural/microchemical changes due to irradiation or thermal ageing.
- Identify suitable homogenization schemes to apply continuum plasticity approaches and evaluate the mechanical behaviour at component level.

Here too model validation/calibration require suitable mechanical characterization experiments, including **micromechanical characterization from specimens at single grain scale** (nanoindentation, micro-pillars, ...) with different lattice orientations, coupled with microstructural analysis, as well as mechanical tests addressing **uni-axial vs multi-axial load, cyclic load, relaxation, load sequence, non-proportional loading**, etc. Micromechanical characterization techniques are often the only ones applicable to charged particle irradiated specimens, due to the limited penetration of these particles.

Figure 6 illustrates in a schematic way the idea of the ICME approach, as well as the modelling and experimental techniques involved. The relevant infratechnology here introduced corresponds to the second

³¹ E.g. high resolution TEM (HRTEM), or scanning TEM (STEM), coupled to energy dispersive X-ray (EDX) or electron energy loss spectroscopy (EELS), to analyse identify defects and also measure the local chemical composition.

line in the warp of Figure 4, defined as “Advanced modelling and microstructural characterization for predictive capability”, which can be concisely referred to as **modelling**.

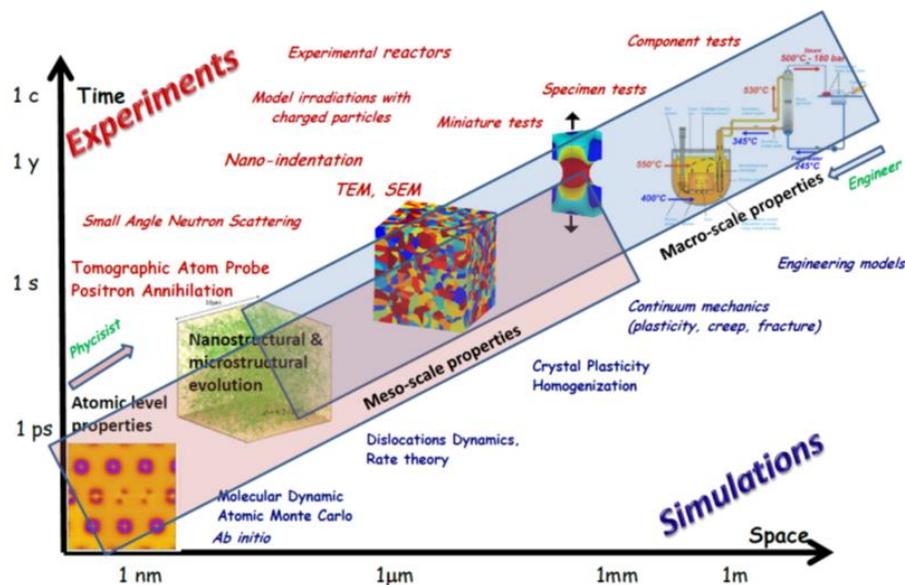


Figure 6: Modelling and experimental characterisation techniques at various time and length scales involved in the ICME approach, used to elucidate the basic mechanisms of nuclear material behaviour and ageing, and their coupling.

2.3.3 Development of advanced materials

The milder action of one or another material degradation factor depending on the component is generally a positive fact, that allows designers to reduce the requirements on materials for specific parts of the plant. However, for some components or parts of components it may happen that no design-driven ageing mitigation strategy can be identified that guarantees their fitness for purpose over a sufficiently long timespan. This means that the availability of the component will not allow an affordable and/or fully safe functioning of the system. In this case there is a **strong push to identify materials with better properties or to suitably modify and protect those that are available to improve their properties, or even to develop altogether new materials.**

This applies specifically to commercial GenIV reactors, because it is very unlikely that the high fuel burnups and operating temperatures targeted in these systems, in contact with non-conventional coolants, can be fully achieved with existing materials, under the overarching principle of safety. It is well-known that prototypes and demonstrators will operate at conditions that are not optimal in terms of efficiency and waste minimisation, at smaller scale and power than eventual commercial reactors, because this is the only way to be compatible with the conditions that existing materials can tolerate, based on existing qualification data and return of experience. **To ensure the commercial deployment of truly GenIV systems with a design lifetime of 60 years and sustaining high fuel burn-ups and transmutation, materials of better performance are needed.**

This requires a significant research and development effort towards: (i) the identification of effective materials solutions to mitigate the consequences of the harsh conditions that are met in the reactors; (ii) the improvement of the properties of existing materials by identifying more suitable fabrication or treatment processes; and (iii) in some cases even the elaboration of completely new materials. It is quite clear that the development of materials implies not only the elaboration of the materials architecture, production or treatment route and identification of reliable joining techniques, but also the qualification of the result of this effort via exposure to suitable conditions, very much in the same way as in the case of existing materials to be qualified for the prototypes. The difference is that the first goal here is not to define correlations and rules allowing the design, licensing and construction of a reactor, but mainly to *screen amongst different possibilities*, to identify the most suitable one(s) Eventually, the full qualification effort to define design rules will be needed, if the new material is meant to be used for the construction of a reactor, but this step will be taken only for a very reduced number of materials, that emerged from the screening selection. The relevant infratechnology here described corresponds to the third line in the warp of Figure 4, defined as “design, manufacturing and processing of

3. Nuclear materials research agenda

This section describes the research issues to be addressed in terms of pre-normative materials research and qualification, advanced materials modelling and characterisation, and development and screening of advanced materials, to provide system designers and component manufacturers with suitable qualified materials and relevant design rules/fuel performance codes, that will allow the licensing and construction of GenIV reactor prototypes and demonstrators, FOAK and eventually commercially plants. It is divided in two parts, corresponding to the two columns that constitute the "hoof" of the "fabric" in Figure 4, namely structural materials (section 3.1) and fuel (section 3.2).

3.1 Structural materials

The following three sections address the different issues (boxes) that enter the three blocks defined by the three infratechnologies of Figure 4, for what concerns structural materials.

3.1.1 Pre-normative research: qualification, test procedures, design rules

This section addresses the development of the materials data and tools in support of the design, licensing, and construction of the ESNII prototypes and demonstrators, also with a view towards the commercial reactors to be deployed from 2040. . Pre-normative research has been a key objective in the FP7 projects MATTER and MATISSE, and is the key focus in the H2020 project GEMMA and in ongoing JPNM Pilot Projects.

An especially delicate issue in this context is the **extension of the operational life of non-replaceable components from 40 to at least 60 years as a general Gen IV requirement**. All relevant slow long-term degradation processes need to be accounted for, especially, high-temperature (creep, fatigue, thermal ageing), but also corrosion and low-dose-rate long-term irradiation. A very fundamental issue is how to measure material properties representative for long-term operation. This is a tremendous challenge shared by all Gen IV concepts, for which design and assessment methodologies need to be developed and translated into codes and standards. As a rule-of-thumb for creep extrapolation a factor 3 is considered feasible, which would require tests of 20 years duration for reliable 60 years design data. In a future low-carbon energy mix, the nuclear contribution will need to operate in a load-following mode, whereby the reactor components will accumulate more load cycles, which should also be taken into account in a long-term operation perspective. Although the 60-years operational life is a requirement for the commercial deployment, thus a long-term need, it is a short-term R&D need to start in due time, i.e. now, the long-term test programmes. The ESNII prototypes, in particular ASTRID, should incorporate the 60-years operational life in the design.

Most structural integrity issues in metallic components occur at the welds, which in combination with the complexity of assessing welds, infer that **much emphasis in the pre-normative research should be dedicated to welded components**.

The high investment cost is perhaps the largest obstacle for the development and deployment of innovative nuclear reactors. **In addition to demonstrate safety, the codes and standards should also ensure cost efficiency**. Pre-normative research related to reduce cost without reducing safety margins include: reduction of undue conservatism in design rules by applying more advanced assessment procedures; assessment of alternative materials and designs; more accurate and less conservative descriptions of loads; updated materials curves based on additional tests methods as well as extended test conditions. This implies that conceptually the development of codes and standards should be based on a physical understanding of materials degradation processes.

Thus the **pre-normative research includes closely integrated experiments and modelling work at the meso and macroscale**. The "engineering approach" is to start from the large length and time scale whereas the "physicist" starts from the other end. For the pre-normative research the starting point is mainly at the engineering scale, but for the understanding of the degradation mechanisms and include them in the design rules and assessment procedures it is necessary to address the proper length and time scale for the particular degradation mechanism. One example is the extrapolation of accelerated data to operational conditions where the proper relevant degradation mechanism at the relevant length scale must be explicitly taken into account. The end product would then be engineering tools based on a multi-scale informed approach. This section will consistently include engineering modelling approaches, while more physical approaches are the subject of section 3.2.1.

Table 4 lists the main issues for pre-normative research for the key materials to be used for the ESNII prototypes and demonstrators. The 60-years design lifetime is not listed as a specific issue as it corresponds to a combination of several of the issues in the table. The reference structural materials are austenitic steels, in particular 316L and 316L(N), that are planned to be used for the core of all prototypes, but also the high-Ni alloy 800H for out-of-core components in GFR. For the latter application, Ni-base alloys are also reference

materials. Alloy 800H is qualified up to 750°C for the steam generator; for the other high temperature components – turbine blades, hot gas header, intermediate heat exchangers – Ni-base alloys with higher creep rupture strength are required, e.g. Inconel 617, Haynes 230, or Hastalloy-XR.³² Emphasis has been put on 9Cr F/M steels which, although not used for MYRRHA and ALFRED, and only to a limited extent in ASTRID, remain candidate materials for Phase II. For cladding materials the reference material is 15-15Ti austenitic steel.

Table 4: Main issues concerning pre-normative R&D: materials qualification, design rules, assessment & test procedures.

| Main issue | Breakdown in sub-issues | Materials concerned | Techniques/Methods |
|--|--|---|--|
| High temperature behaviour and degradation of metals | Creep, relaxation and cyclic deformation | Austenitic steels (316L, alloy 800), F/M steels (Grade91), Ni-base alloys (Inconel 617, Haynes 230, Hastalloy-XR) | <p><u>Experiments:</u> For <i>long-term operation</i>: Mechanical tests of in-service material, long-term tests, accelerated tests.</p> <p>Basic tests for <i>model calibration</i>: creep, low-cycle fatigue, crack propagation tests. Special emphasis on long hold times.</p> <p>Microstructural analysis to link mechanism-based models to experiments are needed.</p> <p><u>Models:</u> For creep, relaxation and cyclic deformation, emphasis on unified viscoplastic continuum models, especially diffusion creep models (meso-to-macro scale approach), to address in-service conditions and non-isothermal conditions.</p> <p>For damage and crack propagation fracture mechanics and damage, models that integrate creep and fatigue are required, and in the long-term also microstructural evolution. The interpretation of accelerated tests requires mechanism-based models at appropriate length scales.</p> <p>In all cases the models need to be translated into Design Rules or Assessment Procedures.</p> |
| | Creep and creep-fatigue damage and crack propagation | | |
| | Thermal ageing | | |
| Environmental compatibility between coolant and structural materials | Liquid Metal Corrosion and erosion (LMC) | Austenitic steels:316L, 15-15Ti | <p><u>Mechanical tests:</u> slow-strain rate tensile; fracture, fatigue, and creep-fatigue in flowing and stagnant conditions;</p> <p><u>Corrosion tests:</u> Erosion and corrosion (oxidation and solution tests) in flowing conditions</p> <p><u>Qualification tests</u> (mechanical and corrosion) for 316SS and welded components.</p> <p><u>Accelerated tests</u> to map bounding conditions.</p> <p>Emphasis on <u>long-term tests</u>; A very careful documented control and monitoring of the test conditions (in particular oxygen control) is required for all tests.</p> |
| | Liquid Metal Embrittlement in HLM (LME) | (P91 and AFA steels and coatings) | |
| | Compatibility with HT He | Austenitic steels 316L and alloy 800, Ni-based alloys (Inconel 617, Haynes 230, Hastalloy-XR), as well as | |

³² There has also been effort to develop special ODS alloys with high temperature strength and stability (MA6000).

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| | | SiC/SiC and AFA steels. | <p>Tests to be complemented by detailed microstructural analysis (e.g. SEM, EBSD, XRD, TEM);</p> <p>Engineering related approaches need to be developed such as path dissolution and film rupture for stress corrosion cracking or purely empirical models. Coupling with models developed in 3.1.2</p> |
| Radiation effects | Low temperature embrittlement & plastic flow localisation | Austenitic and F/M steels (irreplaceable & structural components) | <p>Exposure to irradiation, also including coolant environment.</p> <p>Standard mechanical test in hot cells of neutron irradiated materials (irradiated in test reactors or in-service exposed) and complementary ion/proton irradiation.</p> <p>These tests need to be supported by irradiation models from 3.1.2</p> <p>Transfer the data into reduction factors for material properties.</p> |
| | Long-term/low dose irradiation in environment | | |
| | High dose irradiation swelling and creep | Fuel cladding materials: 15-15-Ti austenitic steel | |
| Assessment of complex loadings | Non-isothermal thermo-mechanical loads | All | <p><u>Experiments:</u></p> <p>Component or specimen tests that simulate thermo-mechanical loads (e.g. thermal shocks), variable amplitude tests for fatigue; tensile high strain rate tests, multi-axial tests, e.g. cruciform specimen for biaxial loading.</p> <p><u>Modelling:</u></p> <p>Finite element models of complex load cases and simplification to translate into design rule load cases.</p> |
| | Complex stress distributions (3D, gradients, stress concentrators, variable amplitude) | | |
| | Load transients and beyond design conditions | | |
| Integrity and qualification of weldments and welded components | Residual stresses | <p>All welds:</p> <ul style="list-style-type: none"> • austenitic, • ferritic-martensitic • Ni-based alloys and • dissimilar metal welds | <p><u>Experiments:</u></p> <p>Residual stress measurements by neutron diffraction and simpler but less accurate methods such as X-ray diffraction, contour method.</p> <p>Standard test for welded specimens (tensile, fracture toughness tests, low-cycle and high-cycle fatigue) ;</p> <p>Characterization of the different regions of a weld (small punch, nano and micro indentation)</p> <p>Mock-up test of welded component for validation.</p> <p><u>Microstructural analysis</u> (SEM, TEM, XRD, EDX)</p> <p><u>Modelling:</u></p> |
| | Weld procedures | | |
| | Degradation modes and defect assessment | | |
| | Compatibility with HLM | | |

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| | | | Simulation of weld process and post-weld heat treatment to compute residual stresses by FEM taking into account phase-transformation, thermos-mechanical processes. Structural integrity assessment (defect assessment crack propagation, damage) of welded specimens and components by FEM. Translation of structural analysis assessment into Design Rules. |
| Sub-size and miniature specimen test standardization | Thin-walled cladding tubes | Fuel cladding material, 15-15Ti | Experiments: Various fuel cladding tests (internal pressure, ring-compression, small-punch, conde mandrel) with emphasis on hot-cell tests; Small-punch test for tensile and creep properties; Nano-indentation, micro-pillar tests for tensile properties; miniature specimen fracture and fatigue tests. Modelling: Test to be complemented with finite element analyses, and meso-scale models (dislocation dynamics and crystal plasticity) |
| | Small Punch test | All | |
| | Micro-pillar tests and nano-indentation | All | |
| Component and material health monitoring | | All | Patterns of response of material to NDE techniques as part of codification Exploration of possibility of lifetime estimation based on NDE in view of online monitoring |

• **High temperature behaviour and degradation of metals**

Creep effects need to be taken into account above the negligible creep temperature, which is 470, 425 and 375°C, for Alloy 800, 316SS and F/M steels, respectively. Many fast reactor components operate in the creep regime where **creep, creep-fatigue** and **thermal ageing** limit the operational life of components. These mechanisms are also crucial for the 60-years design life. High-temperatures must also be considered for accident conditions, where temperature excursion could be well above the normal operational temperature.

For stress-limited "forward"-creep deformation, design rules, assessment procedures and standard tests are well established, based on isothermal creep deformation, creep rupture life and negligible creep curves, as essential design properties. Strain-limited stress relaxation is an important high-temperature mechanism for which models are less developed. **The most important pre-normative research need for pure creep is to extend the material curves to the 60-years operational life.**

Fast reactors work at low pressure, but cyclic thermal loads with a typical temperature variation of 100°C are significant. Thus creep-fatigue, or more specifically **non-isothermal creep-fatigue, will be the dominant mechanism.** Design rules, assessment procedures and material curves are much less developed for creep-fatigue than for creep and stress relaxation, so more accurate models are needed. **This problem is especially serious for F/M steels that undergo significant softening under cyclic loading:** this behaviour, at variance with austenitic steels and Ni-based alloys, prevents a simple adaptation to F/M of existing design rules. Understanding the underlying processes and developing suitable models becomes in this case absolutely essential to establish robust design rules.

Thermal ageing is caused by microstructural evolutions and affects the long-term material properties and, most importantly, reduces the ductility of the material. The microstructural evolution depends both on stresses and temperature and is difficult to predict. In the ASME and RCC-MRx codes thermal ageing is only included as reduction factors for some selected properties such as yield and ultimate strength and fracture toughness, and

only for few materials. **Thermal ageing is mainly an issue for long-term operation.** The development of design rules and assessment procedures that account for the effect of thermal ageing is a tough challenge.

According to the above, we distinguish three cases of **model/procedures development**, as follows.

Creep and cyclic deformation

As regards creep deformation, design rules and assessment procedures need to describe creep and relaxation curves for the reference materials under the operational conditions. The creep mechanism for a given material depends on the stress and the temperatures. Most models and material test data in design codes are restricted to relatively high temperatures where creep is governed by dislocation climb. At the operational conditions the stresses and temperatures are generally lower and creep deformation is mainly by diffusion. Extrapolation of dislocation-climb creep data from accelerated tests to operational conditions where creep is by diffusion will give non-conservative predictions. **For the long-term operation it is necessary to have models and data that incorporate the degradation mechanisms or microstructural evolution itself by addressing the proper length scale.**

In actual components creep and relaxation may occur simultaneously but at different locations and generally under cyclic loading. The goal should therefore be to have unified models that include forward creep and stress relaxation, plastic deformation and creep, and that can simulate the key observations in tests such as cyclic hardening (austenitic steel and Ni-based alloys) and softening (F/M steels), progressive accumulated deformation (ratchetting) and the forward-creep or relaxation during hold-times in stress or strain control respectively. The obvious way forward is to use visco-plastic constitutive models that include kinematic hardening and isotropic hardening/softening, and possibly also other effects such as recovery. Since loads are pre-dominantly thermo-mechanical with a very large variation in the creep rate in the temperature range 450-550°C, the temperature effects need to be taken into account. Visco-plastic models are quite complex, but the largest obstacle for their wider application is the calibration of the parameters through different tests. To upscale from mechanism-based models to engineering-based continuum models may require some homogenization schemes through for instance mean-field theories.

Creep and fatigue damage and crack propagation

A sufficiently good prediction of the visco-plastic deformation is needed to predict the life-time of components in the creep range, but it is also necessary to incorporate the initiation of damage and crack propagation until final failure. In the design codes and assessment procedures creep-fatigue life assessment is based on a simple design rule, the **creep-fatigue interaction diagram**, where the fatigue and creep damage usage factors are computed separately and the interaction is only through the empirical interaction diagram. The predictions are not very accurate, which in the design must be compensated by large safety margins. Models that actually integrate creep and fatigue with proper damage criteria are needed for more reliable predictions. Methods are required to predict whether a given defect in a component may become critical. This demands the application of fracture mechanics models and in particular for crack growth in the creep-fatigue regime more reliable models need to be developed. An issue for both damage initiation and crack propagation is the very long test durations needed to get relevant data for tests representative for temperature, load amplitude and hold-times.

Thermal ageing

It is challenging to include thermal ageing effects in design codes and assessment procedures as it is a long-term effect for which simulation by accelerated tests and extrapolation is not obvious. The first engineering approach would be to apply reduction factors for key tensile and fracture parameters. This requires testing of service-exposed material. Prediction of ageing reduction factors by modelling is difficult as it requires prediction of the microstructural evolution and then to relate it to changes in macroscopic material properties. Models of this type are the subject of section 3.1.2.

Data collection and production

The **key to the development of a methodology for a 60 years reactor operation life is the access to representative long-term material data.** There will never be "enough" data so three different but complementary approaches should be pursued.

- Large test programmes, including long-term tests, were performed in the previous decades, in particular for 316L stainless steel grades and welds. Compilation and assessment of such "historical" data should be the starting point. Unfortunately data management plans were not developed at that time and the data available is rather limited and not so easy to use.
- A second extremely useful approach would be to perform mechanical tests and microstructural analyses of materials from reactor components operated in the past, such as the French sodium fast reactors and the UK AGR reactors, the reactor type which has the largest accumulated operational life in the relevant temperature range (500 – 600° C).
- New test programmes need in any case to be defined and performed. This should include materials and components manufactured in accordance with the most recent specifications in RCC-MRx. The bulk of the

new test should be different accelerated tests, complemented with long term validation tests and to include ageing. The key to the relevance of accelerated tests is that the same degradation mechanism should be activated as for the long-term tests. Mechanistic models are needed to define an optimized test plan, as well as to evaluate the post-test results. Acceleration is not simply a question of increasing the loads and the temperatures. New test procedures need to be developed. For instance creep damage can be accelerated by increasing the stress-triaxiality. In stress relaxation tests the creep rates decrease to in-service relevant levels much faster than in classical uniaxial (forward) creep tests. Stress relaxation could potentially be used as accelerated creep tests provided a visco-plastic model exists that can describe both phenomena.

- ***Environmental compatibility between coolant and structural materials***

HLM-cooled systems

While liquid sodium is relatively mild to steels, it is well known that material in contact with **heavy liquid metals (HLM)**, i.e. liquid lead and lead alloys such as lead-bismuth eutectic (LBE), may degrade severely by **corrosion, dissolution and erosion**. In addition, **liquid metal embrittlement (LME)** is a phenomenon specific to some materials that may affect severely the mechanical performance and determine the failure of components. So, in **HLM-cooled reactors the main issue is to find materials that retain their long-term integrity in contact with the coolant**.

Concerning resistance to corrosion, it is essential to **map, for the selected materials and coolant, the corrosion rate as a function of all the variables involved**, namely temperature, impurity content (especially oxygen), and fluid velocity. It must be kept in mind that in the reactor it will be impossible to guarantee the same level of variable control as in a laboratory test, thus the range of testing should exceed the range of nominal service conditions and emphasise the most penalizing conditions. Corrosion tests need to be conducted in flowing conditions whereas mechanical tests (slow strain rate, fracture, creep, fatigue and fracture toughness tests) can partially be done under stagnant conditions. Importantly, **test procedures and standards are not fully developed for HLM environments so activities in this sense need to be considered**. The resulting mapping should be eventually the basis for the elaboration of design rules.

Three **steps towards design rules and licensing**, valid also for LME, can be identified:

- Demonstrate "immunity", i.e. that testing in environment is equivalent to testing in air;
- If immunity cannot be demonstrated but environmental effects are relatively slow, acceptable and predictable, establish a mitigation programme to ensure long-term integrity based on adaptation of the systems parameters, acceptance criteria and inspection plans to accommodate affordable corrosion allowances.;
- If no solid mitigation programme can be identified, then the mapping of the behaviour in environment of *non-conventional* materials needs to be considered: these may be surface engineered steels (coatings) or alumina forming steels, as described in section 3.1.3 (pages 52 and 59).

Based on an approach of this type it has been seen that F/M steels suffer from severe LME at the operational conditions for both MYRRHA and ALFRED and have therefore been ruled out as structural materials in contact with HLM. In contrast, **austenitic 316 steels seem to be quite resistant to HLM degradation, but this needs to be further demonstrated by an experimental programme that should include welded components and define clear boundaries of operation** (mapping).

The environmental degradation under operational conditions may be quite slow and long-term tests are needed before the effects can be observed. Complementary accelerated tests by more aggressive environment may then be needed. More aggressive environment is also needed to establish boundary conditions. In addition to varying temperature and loads (for mechanical tests), the liquid can also be made more aggressive by changing the chemical composition. The interpretation and use of accelerated tests data will be much more complicated than for high temperature degradation, though. As for the high temperature degradation, accelerated tests need to be complemented with appropriate models and microstructural characterization to ensure consistency of degradation mechanisms. Importantly, for completeness a qualification test programme should also include the simultaneous effect of irradiation and environment (see also next section).

Gas-cooled systems

For applications in GFR, the candidate materials (austenitic and F/M steels as well as Ni-based alloys) need qualification at high temperature (see previous section) in contact with flowing He.

The alloy corrosion behaviour depends to a large extent on the relative impurities level in the He coolant and whether oxidation, carburization or de-carburization occurs. Carburization is linked to loss of ductility whereas decarburization reduces the creep rupture strength. The effect is however much smaller than for LME. Ideally a continuous passivating oxide layer develops which provides corrosion resistance. Due to high dilution of gaseous species, the He-coolant is not in thermodynamic equilibrium. This and the fact that the alloy

composition changes makes the prediction of the conditions for stable oxide formation very difficult. Furthermore, the possibility of environment and high stress synergism on corrosion and crack formation should be also investigated. To understand the corrosion mechanism and their impact on materials properties and how they are affected by alloy and gas composition requires test programme complemented by physical and empirical models. The approach to demonstrate the suitability for candidate materials in high-temperature He are similar to the HLM outlined above. However, due to the higher temperatures of GFR, coupling with creep and creep-fatigue becomes more important. Moreover the number of candidate materials with different features is higher than for LFR, which means that a screening is needed. For example the alloy 800H is a qualified material, but it is relatively weak in strength at high temperature (above 750°C), exhibiting also significantly higher coefficient of thermal expansion compared to Inconel 617 and Haynes 230. This makes this alloy potentially susceptible to thermal fatigue. Therefore a significant effort must be made to select other corresponding materials, i.e. Inconel 617, and to develop data to support ASME code cases in order to extend its operating range.

- **Irradiation effects**

For irradiation effects it is necessary to distinguish between fuel cladding, necessarily exposed to high irradiation dose rates, and structural components that are exposed to very low irradiation fluxes. In the RCC-MRx code, radiation of structural materials is taken into account by very basic design rules for tensile strength, uniform elongation, fracture toughness and the elastic allowable stresses. There are two essential border lines: **negligible irradiation curve** that define conditions for absence of significant irradiation effects, and **maximum allowable irradiation**. If negligible irradiation conditions are fulfilled, then radiation effects can be ignored. For values in-between irradiation effects need to be taken into account. In the RCC-MRx irradiation effects are taken into account.

Nuclear reactors are designed so that all structural components should remain in the negligible irradiation regime. The irradiation data, however, are only provided for a small number of materials, and even when known the negligible irradiation curve and maximum allowable irradiation are based on a rather limited number of tests of irradiated materials. In RCC-MRx negligible and maximum allowable irradiation damage in the temperature range 425-550°C for 316L(N) are 2 and 24 dpa, respectively. These values are based on **irradiation hardening and embrittlement**. There is some concern that **with 60-years operational life low dose may have significant effects due the helium production induced by transmutation**. For fuel claddings the main concern is irradiation creep and swelling, the latter also related to He production.

In the case of **F/M steels**, not yet fully codified in RCC-MRx, **low temperature (<350°C) radiation embrittlement** may lead to a significant shift of the ductile-brittle transition temperature (DBTT) already after fractions of dpa, and potentially severe **loss of elongation due to plastic flow localisation** may ensue. The latter require intense modelling effort to be fully understood, so as to justify a revision and extension of design rules for F/M steels (see section 3.1.2, page 46).

Helium is essentially insoluble in metals and the entrapped helium from transmutation tends to precipitate as nano-scale bubbles at grain boundaries. At elevated temperatures, helium bubbles grow rapidly under the influence of both temperature and stress. The growth of GB helium bubbles may result in the weakening of the grain boundaries and intergranular fracture, leading to severe embrittlement in the materials.

Another important aspect to be quantified under irradiation is possible influence on the response of materials to coolant exposure. Synergic effects may exist that exacerbate corrosion, dissolution, or LME. These effects need to be quantified and taken into account, possibly at the level of design codes and certainly at the level of component lifetime management.

In summary, for structural materials pre-normative research may be needed to:

- Expand the class of materials for which irradiation data are provided;
- Refine the irradiation rules by design by analysis, for instance by support of mechanistic models for plastic flow localization and irradiation embrittlement;
- Quantify the effects of helium production long-term low dose exposure;
- Quantify the potentially synergic effects between irradiation and corrosion.

How to translate test data and mechanism-based irradiation models to engineering tools remains in several cases an open question.

- **Assessment of complex loadings**

The structural components in a nuclear reactor are exposed to complex loadings. A good description of the loads and the computation of the stress and strain distributions, which are directly related to the constitutive models, are necessary to predict safety margins, evolution of damage and remaining lifetime of components. Typical complex loadings include:

- Three-dimensional stress distributions.
- Non-isothermal thermo-mechanical loads. **These are caused by temperature variations on the surface and the constraint imposed by the component. Temperature gradients give large variations stresses and strains through the component wall thickness that depend on frequency and amplitude of the thermal loads.**
- Load transients and beyond design conditions. **These are characterized by load transients and temperature excursions.**

To address complex loadings correctly the following **pre-normative activities** are required.

- *Development of constitutive models and governing equations that include temperature variations, strain rate and other dynamic effect dependence and constraint effects. Their calibration requires specific material characterization test programmes.*
- *Transferability between specimen and component tests. All standard tests, which are the basis for all constitutive models, are designed to give simple and well defined loadings. Tests that include complex loading effects in components, i.e. **component tests**, are needed to demonstrate transferability between standard tests data and component assessment. Such tests include for instance bi-axial tests and thermal shock tests.*
- **Integrity and qualification of weldments and welded components**

The operating experience of all nuclear reactor types clearly demonstrates that welds are the weak spots in metallic components. Testing and assessment of welds and welded components must thus be an integrating part of pre-normative research programmes for GenIV conditions.

Welds are very complex as they include at least two materials and the welded microstructure will depend on the energy supplied during welding, thus linked with process. Moreover the microstructure and the local material properties vary across the weld.

Welds will always contain defects, the propagation of which will determine the lifetime of the welded component: **defect assessment through inspection and non-destructive examination can reduce costs significantly and increase safety** (see component and material health monitoring at page 42).³³

Welding will also introduce internal residual stresses. These have been the cause of many in-service failures and thus cannot be ignored by designers. Informed design of **post-weld heat treatment to minimize residual stress and optimize properties and microstructure is especially important.** Another problem is that it is notoriously **difficult to quantify residual stresses in components by experiments or modelling.** Residual stress measurements in welded components by neutron diffraction as well as other simpler methods need to be pursued. In parallel models that predict residual stresses with higher predictability by simulating the physical processes need also to be further developed (see section 3.1.2, at page 49).

Since the properties and the behaviour of welded components depend strongly on the welding process it is essential that welded components in test programmes are manufactured in accordance with code requirements, which for the GenIV reactors correspond to RCC-MRx.

Other points:

- Novel welding technologies such as electron beam, friction stir or laser have considerable promise, but must be thoroughly understood and characterized before use by pre-normative research.
- Design rules are based on weld factors for creep and fatigue. It should be verified if the same weld factors can be used for creep-fatigue.
- Weldment design codes must consider through-life degradation in an informed manner, based upon service experience and advanced assessment procedures. It is not possible to test for 60 years prior to entering service, long-term exposure to aggressive environments or neutron irradiation, so physically based mechanistic understanding of degradation and failure mechanisms for all regions of the weld is vital, to allow prediction of service exposed behaviour.

- **Sub-size and miniature specimen test standardization**

Tests methods and associated procedures and standards for very small volumes of material using small and miniature specimens become essential for:

- Neutron irradiated materials available in limited quantity, or in order to minimize the radioactivity;

³³ Defect assessment is included in Appendix 16 of RCC-MRx and at greater depth in the R6 and R6 defect assessment codes.

- Very thin material layer of affected by charged particle irradiation (typically a few μm);
- Service-exposed materials, i.e. tested in service during inspections;
- Thin-walled cladding tubes;
- Evaluation of local properties, for instance heat affected zone of a weld (mm scale), coatings, single crystals (μm scale), or graded functional materials.

Specimens involved range from those for sub-sized standard and less conventional miniature tests, complementary to the standard specimen and component tests, to truly miniaturized samples:

- **Miniaturised specimens for standard mechanical property assessment.** This type of testing involves two types of issues. Firstly, testing setup and specimen preparation need to be modified to adapt them for small specimens. Secondly, modified evaluation procedures have to be assessed and validated by comparison with larger specimen data, for full transferability of test results. In particular, the issue as to whether these specimens can be used to predict real component behaviour is addressed. Validation of specimen geometry and testing procedure should be undertaken by inter-laboratory exercises as a first step to achieving the standardization of the procedures. One final goal is the determination of the fracture toughness and crack growth rate.
- **Thin-walled cladding tube tests.** This can include tests of different sizes such as small-punch, ring-compression, pressure and cone-mandrel. Since fuel-claddings become highly irradiated, focus should be on testing irradiated material in hot cell conditions. The different tests are complementary, so there is a need to identify the strengths and limitations of each method, and to provide recommendations on procedures and then develop standards on how to perform the tests and evaluate the data. In particular, the **small punch** test is now becoming widespread, also for other applications, due to its relative simplicity preparing specimen and performing tests, but the evaluation of the test result is complex and how to extract material properties is not obvious.
- **Micro- and nano mechanical testing methods.** These include nano-indentation, as well as compression, tension and bending tests on FIB-produced micropillars, where in-situ approaches combine mechanical testing with advanced imaging characterization. Such methods allow single grain properties needed for crystal plasticity to be measured. Testing of bi-crystal specimens can be used to study grain boundary strengths. Nano-indentation allow local hardness variation to be mapped, as tool to understand dislocation dynamics. They all can be applied to both neutron- and ion-irradiated materials and may foster the exploitation of ion irradiation as a surrogate of neutron irradiation. Research is needed to i) further develop high-temperature procedures for nano-hardness measurements, and ii) elaborate test and evaluation procedures for micro-pillar compression tests, including identification of material properties that can be extracted.

There are some restrictions to the applicability of miniature test methods. Creep properties have been studied by small punch test, but restricted to high creep rates. The possibility to derive fracture and in particular fatigue properties is quite limited. The methods are presently mainly aimed for screening and ranking of materials. To what extent design specific material properties (e.g. tensile curve, creep curve, fracture toughness) used in design rules or design by analysis can be extracted is an open issue.

An exception are sub-sized tensile specimens, from which tensile properties, except total elongation, can be accurately obtained: except for total elongation, sub-size samples are already included in ASTM and ISO standards. The use of sub-size specimen for Charpy, fatigue, creep and fracture mechanics tests (fracture toughness and crack growth), in contrast, still needs to be included in the standards.

In summary for the miniature tests there is a need to review in depth the test methods in terms of strength, weaknesses and restrictions to provide material data. This should be followed by the development of common procedures and standardization of these tests. For the sub-sized tests the emphasis should be on fracture and fatigue properties. The standardization work must go through the involvement of national and international standard committees with participation of main testing laboratories and designers. In the case of small punch tests, some activities already started in Europe (ECISS) and USA (ASTM), mainly with international inter-laboratory studies.

- **Component and material health monitoring**

Non-destructive examination (NDE) of components is crucial to verify that components are performing according to expectations and to detect well ahead of time cracks that potentially could lead to failure. However, it is not obvious how to deduce precise information on microstructural evolution and associated changes in mechanical properties from NDE. The approach today is thus to monitor the material degradation by analysing surveillance specimen exposed to in-service conditions by mechanical tests and, if required, microstructural analysis.

Typical methods used for NDE are ultrasounds, X-rays, Foucault or eddy currents and thermoelectric power. Ideally it would be desirable to be able to deduce from these techniques not only information about cracks, but also about how the material has deformed and its microstructure has evolved, possibly identifying signatures that correspond to situations that deserve attention. The current industrial tendency is towards online materials monitoring and lifetime estimation based on NDE. Clearly the GenIV nuclear reactor environment, with components immersed in e.g. liquid metals and exposed to irradiation at high temperature, is not ideal for online component monitoring. However, it is considered that exploring this possibility starting from non-destructive tests for the controllability of materials would provide a clear added value in terms of reduction of conservativeness in design coupled to higher safety standards.

3.1.2 Advanced structural materials modelling and characterization

The issues to be addressed with an ICME approach are necessarily strongly linked to those addressed in pre-normative research, as they correspond to the main degradation processes that affect structural materials for GenIV systems. Since the early 2000, several European projects have addressed from a modelling point of view material degradation processes connected to irradiation, such as radiation hardening and embrittlement (in bainitic, austenitic and ferritic/martensitic steels), as well as, to a certain extent, plastic flow localisation and irradiation assisted stress corrosion cracking in water, namely FP6/PERFECT,³⁴ FP7/GETMAT,³⁵ FP7/PERFORM60,³⁶ FP7/MatISSE.³⁷ Some of the relevant activities currently continue in the H2020/SOTERIA project.³⁸ Issues connected with the modelling of the high temperature behaviour of F/M steels have also been partially addressed in FP7/MATTER³⁹ and FP7/MatISSE,³⁷ while the modelling of oxidation and dissolution, as well as prolonged irradiation, in austenitic steels will be addressed in H2020/GEMMA. In general, several modelling activities are included in EERA JPNM pilot projects. All these projects provide solid bases on which to progress further, both in terms of modelling techniques that have been developed and results obtained, as well as in terms of experimental work, performed with a view to understanding mechanisms and for model validation. However, the development of models necessarily proceeds from simpler to more complex materials and issues, therefore there is still room for the improvement of existing models and there are still several open issues to be addressed. The most important ones are listed in Table 5: *Main issues concerning structural materials advanced modelling and characterization*. and described in some detail in the following subsections.

Table 5: *Main issues concerning structural materials advanced modelling and characterization.*

| Main issue | Breakdown in sub-issues | Main materials concerned | Techniques/Methods |
|---|---|---|--|
| Microstructural and microchemical evolution | Formation of radiation hardening and embrittling microstructural features (low temperature) | F/M (possibly also ODS) and austenitic steels, ceramics | Molecular statics and dynamics using either electronic structure techniques or interatomic potentials for energy calculation, as well as atomistic kinetic Monte Carlo methods, as atomistic models to study stability and mobility of radiation defects, or phase stability. Object kinetic Monte Carlo, or rate theory based cluster dynamics, or to some extent phase field to develop microstructural/microchemical evolution models. |
| | Formation of voids and onset of swelling (high temperature/dose) | | |
| | Radiation-induced segregation/precipitation at extended defects | | |
| | Microstructural evolution under load in relation with irradiation creep | | |

³⁴ FP6 IP PERFECT Project: Prediction of Irradiation Damage Effects in Reactor Components, Journal of Nuclear Materials – Special Issue, vol. 406, issue 1 (2010).

³⁵ <http://www.iaea.org/inis/collection/NCLCollectionStore/Public/46/040/46040904.pdf?r=1>

³⁶ <http://www.sciencedirect.com/science/article/pii/S0029549311001804?via%3Dihub>;

<http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleid=1628620>;

<https://www.openaire.eu/search/project?projectId=corda::190daa7e4bf2ff682991a1c9e23ccfd2>.

³⁷ www.fp7-matisse.eu; <http://www.fp7-matisse.eu/wp-content/uploads/2015/12/MatISSE-2015-GEN-IV-materials-Holmstrom.pdf>

³⁸ www.soteria-project.eu

³⁹ <http://www.eera-jpnm.eu/?q=jpnm&sq=nboard>;

<http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleid=1627574>

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| | Thermal ageing and subsequent precipitation/segregation | | Electron microscopy, atom probe, scattering techniques, positron annihilation, etc used in a combined way to obtain complete description. Specific computational techniques to simulate the response of experimental techniques. |
| | Oxide formation/dissolution in ODS during fabrication, thermal ageing and irradiation | ODS alloys | |
| | Correct interpretation of microstructural examination technique signals and raw data | Any | |
| Mechanical behaviour after and under irradiation (low temperature) | Radiation hardening: flow behaviour | F/M and austenitic steels | Dislocation dynamics models for single crystal behaviour, informed to atomistic models, to deduce constitutive laws. (Strain gradient) crystal plasticity models for aggregates. Homogenisation techniques and continuum mechanics for the component scale. Wide range of mechanical testing, from micropillars to tensile and impact, for model validation/calibration. |
| | Plastic flow localisation | | |
| | Irradiation creep | | |
| Mechanical behaviour at high temperature: | Cyclic plasticity (softening and hardening) | Austenitic and F/M steels | Different models at different length-scales are needed for a full description: from dislocation based models to continuum mechanics. The techniques and methods are the same as in the previous case, but their use is more challenging because of the number and complexity of possible mechanisms, involving not only dislocations but also grain boundaries, and complex thermos-mechanical loading conditions. The up-scaling includes crystal plasticity and visco-plastic models. |
| | Thermal creep | | |
| | Creep-fatigue interaction | | |
| Fracture mechanics | Crack initiation | Austenitic and F/M steels | (Visco)-plastic models including damage criteria; classical fracture mechanics governed by crack tip parameters; local approaches where the fracture processes are explicitly modelled |
| | Crack propagation | | |
| Compatibility with coolants & coolant chemistry | High temperature oxidation/corrosion | F/M and austenitic steels | Atomistic and thermodynamic models; phase field models; extensive microstructural characterization, especially electron microscopy |
| | HLM dissolution corrosion of steels | | |
| | Liquid metal embrittlement | F/M steels | |
| Properties of composite/ceramic materials depending on microstructure/architecture | Tomography of composite and correlation with their macroscopic properties (mechanical, thermal, ...) | SiC/SiC, Max phases, other | In addition to known atomistic and microstructural/microchemical evolution models as for steels, larger scale models that take into account the architecture of these materials need to be identified and developed |
| | Development of models for thermal and mechanical composite behaviour | | |

| | | | |
|--|--|---|--|
| Development of methodology to perform ion/electron irradiation experiments | Design of ion irradiation experiments and interpretation of microstructural data | Model materials, to extend to all | Essentially the same techniques as for microstructural/microchemical evolution, but specifically targeted to charged particle irradiation issues |
| | Use and interpretation of micromechanical techniques (e.g. nanoindentation) | Model materials, to extend to all | Specific simulation tools combining atomistic to continuum descriptions to be developed for the simulation of micromechanical techniques |
| Other issues | Residual stresses after welding Modes of deformation in steels | Austenitic and F/M steels ODS steels | In principle a full suite of codes and methods, through all scales, should be deployed to simulate what happens in the welding process or under deformation in ODS steels. Neither of them exists to date. |

- **Microstructural/microchemical evolution**

Irradiation produces damage in materials through processes of interaction between energetic subatomic or atomic particles with the atoms and molecules that form the target. Therefore, in the case of nuclear materials, that have irradiation as their most specific origin of degradation, **large effort needs to be devoted to the development of models that describe the material at the microscopic submicroscopic and atomic scale**, accompanied by relevant modelling-oriented experiments, as described summarily in section 2.3.2.

Modelling the microstructural and microchemical evolution under irradiation is essential in order to understand, among others:

- The kinetics of formation of **microstructural features that are responsible for radiation-induced hardening and embrittlement** of steels, and metallic materials in general, **including loss of elongation due to plastic flow localisation**, at temperatures $<0.3-0.4T_m$, (where T_m is the melting point in K). These phenomena are often the main lifetime limiting factors for nuclear components and affect materials already after irradiation to very low dpa.
- The kinetics of **formation of voids and the onset of radiation-induced swelling** of steels, and metallic materials in general, at intermediate temperature ($>0.3-0.4T_m$), above a threshold dpa dose, and the steady-state swelling rate above such dose. Predicting the **threshold dose for steady-state swelling onset is crucial**, because this dose mainly limits the lifetime of cladding materials, thereby affecting the maximum burnup that can be reached. This dose is very challenging to predict: it may vary between a few to hundreds of dpa, depending on several variables, like temperature, dose-rate, material type and composition, as well as material history.
- **Segregation and/or precipitation of chemical species, including transmutants like He, at extended defects** (dislocations, grain boundaries) induced by irradiation or thermal ageing in steels, and metallic alloys in general. These processes may significantly change the property of the materials, influencing their response in terms of mechanical properties (hardening, loss of grain boundary strength and promotion of intergranular fracture, ...), but also of reduced resistance to corrosion and oxidation in contact with coolants. Irradiation environments may trigger segregation and precipitation also against thermodynamic equilibrium, making them difficult to predict.
- The **stability and size distribution evolution of carbides and oxides** in stabilized or strengthened steels, and metallic alloys in general, during material fabrication, thermal treatment, thermal ageing, and/or irradiation. Obtaining and maintaining under operation a microstructure characterized by fine and uniform dispersions of particles is indeed key to guarantee the integrity of components at high temperature for sufficiently long time.

Although significant advances have been made in this direction in the last couple of decades, multiple challenges remain to be addressed in order to develop reliable ICME models that provide truly predictive capability. These essentially aim at being able to **describe, with a single tool, the evolution of both the microstructure (redistribution of lattice defects) and the microchemistry (redistribution of chemical species)**, be this based on object kinetic Monte Carlo, cluster dynamics or even phase field approaches. Among others, the following modelling developments need to be pursued:

- Develop **atomistic models allowing multi-million atom simulations for chemically complex systems** such as for example steels. While ideally interatomic potentials fitted to electronic structure calculations

with density functional theory (DFT) would be desirable, for many applications rigid lattice models based on cluster expansion concepts may be comparatively simpler to fit to DFT.

- Develop methods to simulate **radiation damage production in collision cascades** not only in pure elements or simple compounds, but also for complex alloys, **including all involved interactions**, e.g. phonon-electron.
- **Include the effect of applied loads and stress-strain states** as variables in microstructural and microchemical evolution models. This has a twofold application: on the one hand this is needed to describe the influence of stresses on the mobility and stability of radiation defects; on the other it is needed in order to describe correctly the removal of radiation defects at sinks such as dislocations and grain boundaries, including preferential segregation of chemical species, as well as to describe processes such as heterogeneous nucleation of precipitates.
- Use DFT and other DFT-based atomistic models, e.g. based on cluster expansion approaches, to **go beyond and improve models based on the Calphad approach**, for a full physically-based prediction of the phase diagrams for complex alloys and compounds, including kinetics of phase separation.
- Find suitable **compromises between atomistic details and larger scale descriptions**, in microstructural and microchemical evolution models, **to enable high irradiation doses and temperatures in sufficiently large volumes to be modelled**, given that current models have strong limitations in this sense.

Machine learning schemes based on artificial intelligence, e.g. artificial neural networks, could be of valuable help to address the modelling problems related with high chemical complexity.

- ***Mechanical behaviour after and under irradiation***

Microstructural and microchemical evolution models and the corresponding understanding should be used as **input to move to larger scales** and address, for example, the **prediction of the subsequent plastic behaviour**, particularly in tensile tests, as a function of materials type and composition, dose received, irradiation and test temperatures, and also applied strain-rate, as summarily described in section 2.3.2.

The issues to be addressed are, in particular:

- Prediction of **radiation hardening** (increase of yield strength and flow behaviour) **and embrittlement** (reduction of elongation) after irradiation below $0.3-0.4T_m$, due to the impediment to dislocation motion when load is applied, caused by radiation-produced defects.
- Onset of **plastic flow localization**, also after irradiation below $0.3-0.4T_m$, with subsequent **loss of engineering work-hardening and uniform elongation**, and its impact on the overall mechanical behaviour: this phenomenon is intimately connected with radiation hardening, but it is strongly influenced by the nature of the defects that irradiation produces and how they interact with dislocations (absorbable versus shearable). This radiation-specific effect needs to be correctly accounted for if appropriate design rules need to be developed, especially for F/M steels.
- Deformation at constant load due to **irradiation creep**, which occurs in a wide range of temperatures, including low temperatures, and is controlled by point-defect diffusion and unbalanced elimination of vacancies and self-interstitials at the dislocations, therefore differing substantially from thermal creep.

Model developments are necessary to address these issues, as described in section 2.3.2, essentially aiming at **describing in a single framework the kinetic evolution of the irradiation-induced and the dislocation microstructures, to deduce constitutive laws and devise homogenization schemes for continuum mechanics approaches applied to components**.

- As for microstructure/microchemical evolution models, **reliable atomistic descriptions of chemically complex systems applicable to multi-million system atoms are necessary to describe the interaction between dislocations and grain boundaries or radiation defects**, if this interaction has to be representative of real situations. In this case, however, rigid lattice models based on cluster expansion are of no use and interatomic potentials remain to date the only tool to simulate these interactions.
- **A methodology to transfer systematically the above understanding to dislocation dynamics models applicable at single-crystal level** in the form of **local rules** needs to be established: tools exist, but their application to a specific material requires extensive *ad hoc* calculations. Challenging remains in particular to introduce chemical effects (e.g. decorated dislocation loops or heterogeneously nucleated precipitates). In the case of (irradiation) creep, the coupling between microstructural and dislocation evolution needs to be especially effective because *both microstructures evolve simultaneously over time*.

Concerning continuum mechanics, in order to introduce the effect of the presence of boundaries between single crystals in the transfer of stress and strain benefit could be taken from a most widespread use of **strain gradient plasticity**. Moreover, it is not yet completely clear up to what extent current **homogenization**

schemes are able to fully transfer to component level the effects of irradiation or thermal ageing, including **local or localized effects**, such as for example plastic flow localization.

- **Mechanical behaviour at high temperature**

The model developments sketched in the section 2.3.2 are also broadly applicable in the case of high temperature mechanical behaviour, mainly describable as **thermal creep**, associated or not with **cyclic loading**, i.e. **fatigue**, via **creep-fatigue interaction**. However in these cases **visco-plasticity comes into play** to describe the fact that materials at high temperature deform and lose strength also under constant load, through several mechanisms.

Points of attention for models of high temperature mechanical behaviour are the following:

- At high stresses and temperatures thermal creep is controlled by dislocation climb and glide; at lower temperatures and stresses vacancy diffusion (Nabarro-Herring creep) or grain boundary sliding (Coble creep). Since **creep tests are usually at higher temperature or load** than in the operational conditions, **in the laboratory creep is controlled by dislocation climb whereas under operation conditions the creep is governed by diffusion**. So **extrapolation of creep rates from laboratory conditions to operational conditions is generally non-conservative**. The creep failure mechanism are also different. To correctly extrapolate, all mechanisms need to be understood and modelled correctly.
- Crystal plasticity models so far are primarily limited to static tensile loads. In order to address cyclic loading and thus fatigue, however, **time dependent crystal plasticity models need to be developed and calibrated**.
- Separate engineering models are often developed for creep and plasticity but **to understand for instance the coupling between creep and fatigue, unified visco-plastic models are needed**. Examples of these models that include kinematic and isotropic hardening, memory and recovery effects exist and allow features such as cyclic softening and hardening, strain rate effects, etc., to be accommodated. However, the application is limited due to **difficulties in the calibration of model parameters with experiments and in the implementation of models into finite element codes**, requiring large computational resources.

It is important to emphasise that visco-plasticity approaches such as those sketched above assume that continuum mechanics equations, solvable with finite element methods, can be actually written. However, **for several engineering applications connected with, in particular, high temperature operation, and therefore affecting crucially GenIV systems, finite element models may not be usable**. This generally happens because there is no sufficient understanding of the underlying processes to be able to elaborate a conceptual model, which is an essential guide to develop a consistent integrated modelling approach. In these cases, where only or mostly empirical correlations for quantities of engineering interest exist, appropriate microstructural examination together with the (generally semi-empirical) identification of the important variables are of essential use to cast some light onto the fundamental mechanisms and eventually guide an integrated modelling approach. Specifically **in the case of total creep-fatigue life assessment, only semi-analytical engineering models currently exist**, because the underlying processes are not yet clearly identified or are so complex that it is difficult to separate variables and effects. It would be desirable to have more robust, ICME-type models also for these applications, but to date building reliable models of this type for these processes remains an open challenge that needs to be addressed.

- **Fracture mechanics**

All materials contain defects, i.e. microcracks from where cracks can be initiated and then propagate leading to failure. In design codes the assumption is normally that the components are defect free, but **for the structural integrity assessment of components the existence of microcracks needs to be accounted for, especially for welded components**. Existing models need to be improved and further validated, but, importantly, **NDE techniques are key** in order to get a picture of the initial situation of a material in terms of pre-existing (or developed) microcracks.

Neither plastic nor visco-plastic current models explicitly account for the degradation that leads to final failure by fracture. However, by integrating the effect of pre-existing damage, e.g. initiation, growth and coalescence of voids and micro-cracks, into the constitutive continuum mechanics models, the evolution from the virgin state to macroscopic crack initiation should become possible. This implies including criteria for damage on ideally observable quantities (e.g. crack length, crack density).

Crack initiation and propagation are cases of processes for which conceptual models to guide an ICME approach do exist, but the variability of possible situations is so large, the scales simultaneously involved so different, and the phenomena so stochastic in nature, that integrated modelling approaches are inherently

difficult to develop and apply. Here too, appropriate microstructural examination together with the identification of the important variables, although challenging, because they involve the direct observation of loaded materials, are of essential use to cast some light onto the fundamental mechanisms and eventually guide an integrated modelling approach.

The propagation of a given crack can be relatively well described for simple loading conditions. However, in the case of more complex loadings, for instance combination of creep and fatigue, non-proportional loading, thermo-mechanical loads and mixed-mode (combination of normal and shear stresses) the predictability is much more limited.

Fracture mechanics models can be generally separated into two classes: (i) classical fracture mechanics where crack propagation is governed by a crack tip parameter such as stress intensity factor, J-integral or C-integral, and (ii) local approaches where the fracture processes are specifically modelled, e.g. void growth and coalescence for ductile tearing. While the first type of models are the most used and widespread, especially at industrial level, it is desirable that models of the second type should be more and more developed.

- **Coolant compatibility models**

Other complex processes that are crucial in the context of GenIV reactors, but for which no integrated modelling approach is still in place, even though they do have a clear and conceptually known origin at the microscopic scale, are those related with the **interaction between solids and coolants**. The development of suitable models in this case requires not only extensive microstructural examination, but also the identification of appropriate modelling approaches, because the complexity of the relevant physical and above all chemical processes challenges the possibilities offered by existing simulation tools.

The model developments sketched above for microstructural/microchemical evolution under irradiation largely apply here, as well. However **the level of chemical and structural complexity is, in this case, higher, even for the simplest model systems that can be devised** (e.g. Fe or Fe-Cr solid substrate and Pb_{liq}-O as liquid). Models should strike a balance between complexity and level of detail that is not easily identified. Atomistic models can provide clues about mechanisms, but the scales involved and the simultaneous presence of several different phases make the application of purely atomistic approaches very challenging and ultimately probably impossible. Thus atomistic models should inform and match suitable thermodynamic models, for example based on the phase field formalism, describing the **stability of oxide scales**, together with kinetic models describing the processes of **oxidation and dissolution** in, especially, **heavy liquid metals** (Pb_{liq} and its alloys). A proper and fully defined integrated approach is still missing.

Another challenging problem is **liquid metal embrittlement**, the mechanisms of which remain still largely unknown, although they are probably related with the penetration of the liquid in the solid through specific boundaries (grain boundaries, twin boundaries...). Models describing this type of phenomena, and the consequences on the intimate properties of the material are currently not available. Moreover, any low-scale model of this type should then translate into a process that affects the mechanical behaviour in terms of not only loss of elongation in tensile tests, but also loss of fracture toughness and exacerbated fatigue. To date, no modelling tool exists that can properly describe the relevant physical processes. Developing mesoscopic models for these phenomena, rooted in a solid description of atomic-level interactions and mechanisms and linkable to the macroscopic mechanical response, remains therefore an open challenge that needs to be addressed.

- **Models in support of the use of charged particle irradiation**

Neutron irradiations are long, expensive and access to facilities is restricted. In a context of **modelling-oriented experiments** on radiation effects, but also of **screening of different routes to more radiation-resistant materials**, charged particle irradiation (ions, protons, electrons, ...) is a very valuable and affordable tool. However, transferability issues exist between charged particle and neutron irradiation environment. To design and interpret correctly the results of these experiments, it is **essential that suitable models address the specificities of these modes of irradiation, that differ in many respect from neutrons**. Developing models of this type beyond the use of standard binary collision approximation models to draw the profile of ion penetration, remains an open task.

While the scheme sketched above for microstructural/microchemical evolution under irradiation remains broadly valid, **ad hoc solutions need to be devised to take into account the specificity of charged particle irradiation**, namely: limited damage penetration and presence of close-by surface, damage profile and gradient, injected interstitials, effectively pulsed irradiation, possible self-annealing effects, injection of unwanted species ...

Moreover, a strong limitation of charged particle irradiation experiments is that **the quantity of material affected is very small, and yet not uniformly irradiated**. This introduces specific difficulties in the PIE. Advanced micro-specimen fabrication techniques for microstructural characterization, based on the use of

focused ion beams (FIB) to select specific regions, are needed. More importantly, it is impossible to perform a standard evaluation of mechanical properties. It becomes therefore **essential to develop the capability of obtaining meaningful assessments of mechanical properties using alternative, microprobing techniques**, such as nanoindentation, small punch, micropillars, This capability should also go hand-in-hand with suitable mechanical behaviour models, to be able to interpret correctly the results of the experiments. This links with the issue on small specimen testing addressed in section 3.1.1, page 41.

Overall, **the goal is to develop an established and possibly standardized methodology to perform charged particle irradiation and relevant PIE, including mechanical characterization**. This is not expected to be suitable for the qualification of materials for pre-normative purposes, but should at least provide robust support for the development of models, ranging from microstructural evolution to mechanical behaviour. It should also allow different possible materials to be screened in terms of response to irradiation, limiting neutron irradiation to the most promising ones.

- ***Development of standard methodology for model validation***

The validation of models is a complex and costly task that requires specific attention also from a methodological point of view:

- Experiments aimed at validating models need to expose and then characterise materials with a view to identifying mechanisms, thus in them variables need to be separated, model materials are as important as technological materials and the characterization needs to combine several techniques, as only their combination can provide a sufficiently global and complete picture to allow comparison with all the model results.
- The way of applying microstructural examination techniques (TEM, APT, SANS, PAS, ...), as well as micromechanical characterization methods (nanoindentation, micropillars, ...), and the way of analysing the outcome of the examination, is currently not standardized. This creates difficulties when collecting and putting together results from different laboratories. Thus **protocols for the execution of microstructural examination and the analysis of the data need to be established**.
- To facilitate experiment/simulation comparison, **methods that enable the simulation of the signal that the experimental techniques provide**, for instance how a given microstructural feature as simulated is going to appear to APT or TEM or PAS, need to be developed. They exist only in a few cases and their use is not sufficiently widespread.
- Finally, also **protocols according to which the comparison between simulation and experiment should be performed**, in terms of which quantities can be safely compared and using which criteria, need to be identified.

- ***Other issues***

Quite specific issues for which not much has been done in terms of modelling, but that deserve mention, concern the problem of assessing the residual stresses in welded zones and the problem of the deformation and fracture modes of steels in general and specifically ODS steels.

- ***Note on materials of interest and different levels of model development***

It is important to emphasise that **the level of development of advanced models is not equal for all materials of interest for GenIV reactors**. For example, even if austenitic steels are currently the candidate materials for GenIV prototypes and demonstrators, therefore corresponding to the shortest-term needs, **the development of microstructure evolution physical models for austenitic steels is much less mature than for F/M steels**. The reason is a combination of historical factors and of ease to apply the available modelling techniques. The crucial importance of radiation-induced embrittlement of the steels used to fabricate the pressure vessel of current GenII/III nuclear reactors focused a lot of effort on the development of models dedicated to these materials and problem, which have very much in common with ferritic-martensitic steels. Importantly, both reactor pressure vessel steels and F/M steels can be modelled as body-centred-cubic iron alloys, starting from Fe or Fe-C as model material, and adding later the effect of other alloying elements, which is an especially robust way of progressing in model development. The other reason for austenitic steels lagging behind is that they are concentrated alloys that are not stable at 0 K and are characterised, moreover, by a complex magnetic state, thus proving to be quite challenging materials to be modelled using atomic-level calculation and simulation tools. Equally challenging is the connected development of physics-based models to deal with microstructural and, above all, microchemical evolution. In turn, however, plasticity models suitable to address the mechanical behaviour from the mesoscopic to the macroscopic scale are somewhat more evolved in face-centred-cubic crystallographic structures (such as austenitic steels).

This situation has two consequences:

- **The modelling of F/M steels is certainly worth being pursued** not only because these materials remain important in the long-term for GenIV applications, but also because they represent a sort of reference case to be used as example and starting point for the development of models for other, atomistically more complex, materials, including austenitic steels. In addition, problems such as cyclic softening or liquid metal embrittlement seem to be specific of these steels and require to be addressed, in order to guide the qualification of these materials for future reactors.
- **The modelling of austenitic steels needs now to be pushed forward**, in such a way that, by “catching up” with the level of maturity that characterises models for F/M steels on the low-scale side, it can then benefit from the more advanced level that characterises the modelling of face-centred-cubic alloys, when it comes to the scale of dislocations and continuum mechanics, thereby being accelerated.

Since the development of models, in particular at the low-scale, necessarily proceeds from simpler to chemically more complex systems, F/M steels represent also the starting point also to develop models of use to simulate **ODS and/or alumina-forming alloys** (FeCrAl), with a view to optimizing them. In the case of ODS, specific problems need to be addressed in terms of mechanical behaviour concerning their modes of deformation, especially at high temperature, and residual stresses after the fabrication process. On the other hand, high-entropy alloys, for which modelling support could be interesting in order to streamline among the possible combinations (see section 3.1.3), as well as to understand the underlying reasons for their promising properties, are generally face-centred cubic systems.

Also in order to address the problem of **compatibility with liquid metals and other coolants** in connection with the prototypes, while models for austenitic steels are the main goal, including the problem of optimizing alumina-forming austenitic alloys, experience with “simpler” ferritic alloys is expected to help.

Concerning **ceramic materials**, these are often chemically complex systems (e.g. MAX phases), that require specific developments in terms of atomic-level modelling. Of all, SiC and Al₂O₃, are probably the only two systems on which some solid modelling background exists. However, the activities are globally very limited and scattered. More importantly, these materials are hardly ever used in their monolithic form and never as structural materials: in order to improve their mechanical properties, they are used in the form of composites. This opens completely different types of problems for modelling, because the macroscopic behaviour of these materials will be determined much more by the type of architecture of the composite materials and its degradation than by the degradation of the bulk material. Therefore, the monitoring of the behaviour of the architecture becomes of overarching importance, posing problems such as the need to obtain the correct 3D tomography of the material and to model it correctly at the level of continuum mechanics, knowing that in fact the material is not a continuum. Clearly, the role of the interfaces between e.g. fiber and matrix becomes key to correctly simulate the macroscopic behaviour of the material.

3.1.3 Development of advanced structural materials

The development and codification of new materials for nuclear application is a very long process, due to both the overall strict requirements to comply with the safety constraints of the nuclear industry and the requests of the regulators, and the unavoidable need to perform long and costly irradiation campaigns. This prevents the use of optimised materials for the short term ESNII prototypes. Yet, for long term applications in commercial reactors it is important to pursue the development and codification of better performing materials.

The main targets are to **improve high temperature behaviour, minimise radiation effects** and **mitigate environmental degradation**. In this section, R&D needs for six material classes are discussed:

- Ferritic/martensitic steels
- Improved austenitic steels
- SiC/SiC composites
- Refractory metallic alloys
- Modified surface layers for protection from corrosion
- Other perspective materials

For all these materials there are general issues of **fabrication processes to be developed/optimized** and/or **made less expensive**. For some of them **additive manufacturing** may be a way forward that is worth pursuing. There are also general issues of **protection against coolant attack** for those solutions that do not explicitly address this problem. Finally, for all of them different solutions need to be considered and **screening procedures** applied, which are especially costly when irradiation is involved: for some, but not for all, ion irradiation can be a way to go. Table 6 summarizes the main issues related with the different classes of advanced structural materials.

Table 6: Main issues concerning classes of advanced structural materials.

| Main issue | Breakdown in sub-issues | Materials concerned | Techniques/ Methods |
|---|--|--|--|
| Improved austenitic steels | Reduction of susceptibility to swelling | Further stabilized steels | Addition of elements that stabilize the dislocation network (Ni,Si,P, ...) |
| | Improvement of compatibility with HLM | AFA steels | Addition of aluminium with double benefit of alumina-SL formation and NiAl particle strengthening |
| Ferritic/martensitic steels | Improvement of high temperature behaviour | ODS and CSE F/M steels | Dispersion of oxide particles by powder metallurgy (target > 700°C) or tuning of composition and TMT ODS steel fabrication requires optimization/standardization and industrial upscale of fabrication process, and possibly development of alternative fabrication procedures. |
| | Improvement of compatibility with HLM | FeCrAl or modified surface layers | Current F/M as well as ODS or CSE F/M can be coated or their composition modified with addition of Al (FeCrAl, incl. ODS) |
| | Mitigation of low temperature irradiation embrittlement | Reference F/M | (Strategy may be identified via modelling, see section 3.1.2) |
| Development and qualification of SiC/SiC cladding | Degradation under neutron irradiation, especially thermal conductivity | SiC _f /SiC tube with metallic liner ("sandwich" tubes) or ceramic layer | Neutron irradiations & PIE by SEM & TEM (especially PyC layer), thermal conductivity and hermeticity measurements (thermal characterisation methods by steady state or transient measurements) |
| | Corrosion by oxidation, especially in moist oxygen deficient, high temperature conditions (in He, but also HLM Na) | | Corrosion testing in service environment under careful control of the chemistry and varying the concentration of the contaminating species expected. Post-test microstructural characterization Thermodynamic modelling of the SiC environment/coolant system. Testing after application of protective SL |
| | Joining and relevant qualification | | Screening of materials/processes/designs assisted by thermodynamical modelling Realization of sample joints and prototypes Mechanical testing and performance assessment in the the operative conditions Post-test examination by SEM, TEM, XRD |
| | Standardization of mechanical tests and analysis for tubular | | Standardization of testing procedures |

| | | | |
|--|---|---|--|
| | structures of continuous fiber ceramic composites | | Development of standards & design rules |
| Refractory alloys | Development of processing routes | ODS Mo & V-Cr-Ti alloys | Alloy design & synthesis, regain processing skills. Development of welding procedures. |
| | Environmental issues (oxygen corrosion and interstitial impurity embrittlement, especially in HLM) | | Protection with coatings and relevant qualification. |
| | Characterise and improve behaviour towards swelling & radiation embrittlement | | Neutron irradiation at relevant DPA doses, post irradiation characterization by TEM, SEM, mechanical testing. Improve DBTT shift under irradiation. |
| Surface layers for protection against coolant attack | Optimisation of SL composition and deposition method | GESA surface modification on steels | Assessment of the long term behaviour of the system by thermal aging tests & thermodynamic modelling of the coating substrate system, with a view to progressive optimization. |
| | Evaluation of effectiveness of protection against prolongs exposure to HLM, especially high speed erosion | GESA and PLD alumina protected austenitic steels, F/M steels, refractory alloys | Corrosion testing in service and off-normal conditions. Post-test microstructural characterization (XRD, SEM, TEM, Raman, ...) to identify the compounds formed. |
| | Evaluation of performance under irradiation | | Ion and neutron irradiations & post-irradiation microstructural examination. |
| | Mechanical compatibility with the substrate & thermal aging. | | Mechanical testing under relevant conditions: fracture toughness, cyclic loads (thermo-mechanical fatigue, high cycle fatigue, creep and creep rupture) |
| | Welding of steels with modified SL | | Analysis of welded samples under above conditions. |
| Prospective materials | High entropy alloys | | Explore different combinations of elements, test them from the high-temperature mechanical, corrosion resistance and irradiation behaviour viewpoint. |
| | MAX phases | | Explore different compositions (solid solutions) and microstructures to optimize the material for the specific application. Perform relevant qualification. |

• **Improved austenitic steels**

Austenitic steels have been used as cladding and core component structural materials in past fast reactors. Their behaviour is therefore relatively well-known and overall they offer excellent mechanical properties in a wide range of temperatures, including satisfactory creep resistance up to 700°C. The **main shortcoming for the use of austenitic steels as core components is the susceptibility to irradiation void swelling**, probably both an inherent property connected with the fcc crystal structure and also a consequence of the He

production by transmutation through nuclide chains that start with Ni. Swelling leads to unacceptable embrittlement above 3%.

The 15-15 Ti-class of austenitic steels was developed, starting from the 316 formula, specifically to stabilise this type of alloys against swelling. Ti addition with optimized C content leads to precipitation of carbides finely distributed in the matrix and may also contribute to reducing point-defect recombination by enhancing recombination. The fine carbide distribution stabilizes the dislocation microstructure obtained by cold-working against swelling, allowing 90 dpa to be safely reached: a stable high dislocation density acts as sink for point defects, thereby delaying swelling. Further improvements are probably possible, especially with the help of suitably physical models that clearly identify and quantify in detail the mechanisms responsible for swelling (section 3.1.2).

Another issue of austenitic steels is that, despite their generally higher resistance to corrosion than F/M steels, they **may not offer sufficient guarantees of corrosion-resistance in HLM-cooled systems**, especially with a view to pushing up the operating temperature for higher efficiency.

Swelling-resistant austenitic steels

A relatively recent new specification of 15-15Ti, with better swelling resistance, is AIM1, which should push the maximum allowable dose to 115 dpa and is being qualified now as cladding material for ASTRID (target burnup: 110 dpa). For the future, in addition to ODS cladding or while this is being developed, the effect of swelling inhibitors in solid solution should be explored, in order to develop a further optimized composition (AIM2). These are elements like Ni, Si and P (partially optimised in AIM1), which are expected to decrease the kinetic of void formation, but also Nb and V that, by promoting carbide precipitation, should further stabilize the dislocation network. A downside is that an excess of carbides might lead to embrittlement, that needs to be avoided. Ion irradiation is here a very useful tool to be used for the screening among several nuances of composition.

Alumina forming austenitic steels

Recent material research has focused on the development of alumina forming austenitic (AFA) steels with the aim to increase the high temperature creep strength and corrosion/oxidation resistance for high temperature applications. AFA steels may contain Al in the range 4-6 %wt and a maximum of 25 %wt Ni, exhibiting superior oxidation resistance up to 900°C, due to the formation of a protective Al₂O₃ scale rather than the Cr₂O₃ scales that form on conventional stainless steel, while offering a creep strength comparable with some superalloys containing a much higher amount of nickel, thanks to NiAl particle strengthening. Some alumina forming alloys have been tested in molten lead alloys environment showing good corrosion resistance. For this reason they are being considered as one the most promising materials for HLM applications, in particular for the steam generator of the future HLM-cooled plants.

AFA steels cladding tubes can be accordingly considered as promising option for the long term deployment of the LFR, provided that neutron-resistant materials are produced. The materials under development are very similar to the '□' hardened superalloys studied in the context of the US liquid metal fast breeder reactor programme, for which a large amount of neutron irradiation data are available. The lesson learned during the above research is that the use of thermodynamic simulations and ion irradiations could be used for the accelerated development of radiation resistant AFA. Nevertheless their application as fuel cladding materials will require several compositional changes and qualification under neutron irradiation, before converging to a suitable composition.

• *Ferritic/martensitic steels*

Despite their high thermal conductivity and excellent dimensional stability under neutron irradiation, the core applications of F/M steels in ESNII prototypes are currently limited to hexagonal cans in SFR (see Table 2). Their **use as cladding or core material in other systems is presently prevented by:**

- loss of strength at high temperature (T>550°C), including softening under cyclic operation;
- neutron embrittlement at low temperature (T<350°C), including plastic flow localisation with subsequent drastic reduction of uniform elongation;
- susceptibility to liquid metal embrittlement in contact with HLM.

These phenomena differentiate them from austenitic steels and complicate their codification, because design rules cannot be simply an adaptation from those used for austenitic steels. For this reason, even existing F/M steels such as T91 (or Eurofer for fusion) have not yet been fully codified in RCC-MRx.

Nevertheless, in the long term the use of F/M steels is very desirable in a context of optimal use of resources (high-burn up), given that at present these are the only available industrial materials that **can bear the promise of withstanding neutron doses in excess of 150 or even 200 dpa**.

To be able to use F/M steels for GenIV reactors a number of actions are necessary:

- **Improvement of the high temperature behaviour** by producing more thermally stable strengthening microstructures: this is pursued by oxide dispersion strengthening (ODS) using powder metallurgy fabrication techniques (ODS steels), but can also be achieved by applying suitable thermo-mechanical treatments (TMT) to steels produced by conventional routes, and/or by tuning the composition, supported by thermodynamic models (creep strength enhanced, CSE, steels). The target service temperature is up to 650°C by CSE or above 700°C with ODS steels.
- **Improvement of the compatibility with HLM, by investigation the mechanisms responsible for corrosion and, even more importantly, liquid metal embrittlement**, to identify relevant mitigation strategies, most likely requiring surface engineering solutions (coatings or self-protection).
- **Investigation of the mechanisms responsible for the low temperature loss of ductility** and identification of mitigation strategies concerning low temperature embrittlement and loss of uniform elongation as a consequence of slip localisation.⁴⁰

Any solution proposed to mitigate these problems requires not only an effort in terms of improved material development, based on appropriate understanding and models of the underlying ageing processes, but also of qualification and pre-normative research, in view of the relevant codification.

The two proposed alternatives included here to improve high temperature strength of F/M steels, ODS and CSE, should also include the assessment of the potential cyclic softening, low temperature irradiation (slip localization) and compatibility with coolants. Both classes of steels have potential application also in other energy technologies where high temperature and temperature gradient effects are an issue, e.g. concentrated solar power

ODS steels

Increasing the operating temperature as well as the mechanical strength of F/M steels by a fine dispersion of oxide particles in the matrix is considered a viable path towards improved properties for cladding applications. ODS F/M steels offer indeed improved strength at temperatures above 550°C, while keeping the outstanding swelling resistance of the ferritic (or ferritic/martensitic) matrix. ODS cladding tubes have been produced and their qualification is included already in the early phases of the ASTRID SFR.

The homogeneous dispersion of oxide particles in the nanometre range is obtained thanks to a powder metallurgy fabrication procedure, where the atomized steel particles of the desired chemical composition are milled with oxide particles (usually yttrium oxide) by mechanical alloying. Post-processing steps involve degassing, consolidation (by HIP or extrusion) and annealing treatments. The **nano-sized oxide particles in high number density not only improve creep strength, but also act as sinks for irradiation induced defects and thus improve radiation resistance**. This effect is more or less strong depending on the chemical composition of the ODS and the nature of the nano-oxides, which therefore play an important role

The main issue affecting ODS cladding is certainly the **high cost** associated with the multiple fabrication steps needed in a classical powder metallurgy route, which is also a **bottleneck for upscaling to industrial production**. In addition the microstructural details associated with the individual steps of the processing route are not yet well understood: this currently **limits the reproducibility of the quality of the end product**. In particular, the resulting microstructure after tube production is highly anisotropic. Removing this **anisotropy** is not trivial and require the still open development of suitable intermediate thermal treatments leading to controlled recrystallization. So there is a **need to optimise and standardize the fabrication process, guaranteeing reproducibility and industrial scalability**. Possibly, **alternative and cheaper fabrication routes** should also be identified. **Additive manufacturing** presents itself as a promising way forward.

Finally, an open issue is the development of **appropriate welding procedures**. Conventional fusion welding destroys the distribution of the fine oxide particulates which give the steel its strength and creep resistance.

⁴⁰ This point is not addressed here but in section 3.1.2, page 48, because essentially based on model development.

So presently no established welding methodology has been devised, yet, and further research is needed in this direction.

The TRL of ODS steels remains low: the priority here is therefore to improve the production routes, to guarantee reproducibility and industrial production, in partnership with steel-makers (although currently hardly any is equipped for ODS production, especially in Europe), together with analysis of deformation and fracture mechanisms and stability of the microstructure under irradiation and at high temperature (again, physical models are expected to be of use for these purposes, see section 3.1.2).

TMT/composition-tuned creep-strength enhanced F/M steels

A promising alternative or at least complement to the ODS obtained by mechanical alloying are conventionally produced (i.e. by casting in crucible) F/M steels, the **creep performance of which is enhanced by particles (carbide) strengthening, via conventional metallurgical techniques**, based on fine tuning of the chemistry and suitable post-casting thermomechanical treatments. These are referred to as CSE F/M steels. It has been shown that compositional adjustments on conventional 9-12%Cr steels and thermo-mechanical treatment optimization guided by computational thermodynamics produces a better distribution of the reinforcing carbides and a more stable microstructure at high temperature. Notably, for use outside nuclear energy F/M steels operating up to 650°C have been produced, but no nuclear grade steel of this type currently exists.

The current use of these classes of steels for cladding (but a priori also for other core components) requires the elaboration of the corresponding design criteria and assessment procedures, that, in turn, require dedicated testing for data collection and physical modelling, including stability of the optimised microstructure under irradiation. The codification of conventional CSE F/M steels is expected to be simpler than for ODS, that exhibit significantly different deformation and fracture behaviour, by being linked to the more general issue of the elaboration of design rules for existing F/M steels (see section 3.1.1).

Improvement of compatibility with coolants

F/M steels are prone to **corrosion by dissolution and erosion** when in contact with flowing liquid metals. While this problem is shared with other steels and classes of materials, this specific class of steels, as well as many other alloys with body centred structure, is reported to suffer from **liquid metal embrittlement** when exposed to heavy liquid metals.

The problem of corrosion by dissolution and erosion can be faced by applying suitable surface protections (see below in this section) or by modifying the composition of the steel with addition of Al, which would promote the spontaneous formation of a stable aluminium oxide (Al_2O_3) layer, providing a self-healing corrosion resistant steel. Steels of this type, denoted as FeCrAl, which are the equivalent of AFA for F/M steels, exist already and are also industrially produced: they may contain up to 20-30 wt%Cr and between 4 and 7-10 wt% Al. However they still need composition optimisation, in order to minimise side-effects of the addition of Al (decrease of ductility despite fully ferritic microstructure), while guaranteeing the formation of a stable and continuum oxide layer, which requires reducing the content of both Cr (towards the content typical for F/M steels in nuclear applications, i.e. 9-12 wt%) and especially Al. This objective can be reached by adding small quantities of suitable reactive elements, for example Hf or Zr.

LME represents a different type of problem. It corresponds to a **loss of elongation that appears when a specific metal is tested in contact with a specific liquid metal**, with effects also on fracture toughness, fatigue behaviour, and possibly even creep-resistance. A large amount of experimental data from mechanical testing in LBE and, to a smaller extent, in pure lead environment, has been obtained during the last decades and, thanks to the extensive work done, the features of the phenomenon are at least qualitatively delineated. This work has highlighted the need for direct and intimate contact between liquid and solid metal and the conditions under which the susceptibility to LME is especially high. It has also highlighted, however, how difficult it is to rule out that these conditions can be reproduced in operation and the fact that oxide protection is not necessarily a solution, especially in connection with fracture toughness. Austenitic steels seem to be much less affected by this issue, at least below 400°C; yet the use of some type of filler material to reduce the susceptibility to hot cracking may lead to δ -ferrite contents up to 5% in the welds. LME could therefore have a serious impact on the mechanical properties of welded joints of austenitic steels exposed to HLM. The role of unstable austenite having a martensitic transformation under mechanical strain is also a possibility that would render austenitic steels more susceptible to LME.

A **clear understanding** of the mechanisms behind LME in F/M steels and the synergy between LME and irradiation hardening is **still missing**, as well as a model to rationalize the whole body of available experimental data. The lack of understanding based on experimental observations, calls for the use of advanced modelling tools in order to explore hypotheses and mechanisms from a physical standpoint (section 3.1.2). It is widely recognized that the complexity of the phenomenon and to the difficulties in controlling the experimental details make it difficult to discriminate the relevant processes to LME from those that are not. **The identification of a**

proper framework for materials codification and licensing whenever LME is involved is expected to be especially delicate.

- ***SiC_f/SiC composites***

Increasing the temperature of operation beyond 800°C, such as in the case of GFR and VHTR, requires the use of refractory materials that allow access to temperatures well beyond the current limits of the most heat resistant super-alloys. Of all the possible material classes considered in the past, the main and conceptually most advanced candidates are those based on SiC_f/SiC composites, i.e. SiC fibers in a SiC matrix. By sublimating into vapour directly from the solid at 2830°C, this is indeed one of the most temperature resistant materials available on earth. These composites are also proposed in the commercial deployment of the SFR for the realization of the hexagonal cans and the Na gas heat exchanger, as well as for cladding and structural materials for high temperature HLM cooled systems, due to their good corrosion and erosion resistance.

Thin-walled SiC_f/SiC tubes are fabricated using textile methods where fine filaments (~ 10 µm) combined into fibre tows, are woven or braided into a tubular cladding preform and bound into place by an additional bulk SiC matrix, by chemical vapour deposition (CVD), designed to prevent the weave from moving under stress. An interlayer of pyrolytic carbon (PyC) is deposited by chemical vapour infiltration (CVI), acting as solution of continuity between fibers and matrix. The PyC interlayer transfers the mechanical loads within and through the ceramic fiber reinforced composite, allowing the material to flex and twist without catastrophic brittle failure. At the macroscopic level composites of this type emulate a plastic behaviour, but at the microscopic level they accommodate the deformation by micro-cracking damage. **The main weakness of these composites as cladding materials is that, both as produced and especially after deformation, they are no more hermetic and therefore cannot retain the fission products.**

A potential solution to this issue is the “sandwich” concept recently proposed and patented by the French CEA, where a metallic liner is interposed between two structural SiC_f/SiC shells to secure hermeticity against fission gas release. SiC_f/SiC composite thin-walled, small diameter tubes with one closed end and internally lined with a refractory metal are thus the pin type fuel elements envisaged by the present design for the high temperature configuration of the GFR. An alternative design, the triple-layered (triplex) structure, features a fully ceramic component made of high-density monolithic SiC tube to insure the gas tightness and an outer layer made of SiC_f/SiC to prevent brittle fracture.

Various processes produce the same local chemical bond between the silicon and carbon atoms, but different porosity, crystallography (beta or alpha phases) and Si/C stoichiometric ratios. These influence strongly the overall mechanical and thermal behaviour of SiC_f/SiC, as well as its performance under irradiation.

SiC behaviour under irradiation

Nuclear-grade SiC_f/SiC composites have shown to be stable to extremely high irradiation doses after the initial swelling and changes in strength and thermal conductivity have saturated at relatively low irradiation dose (0.1·10²⁵ n·m⁻², E> 0.1 MeV).

The **swelling** behaviour of high purity beta SiC with stoichiometric composition (Si/C = 1) is relatively well-known in the temperature range of interest (400°C-1600°C): it decreases with increasing temperature to a minimum at ~1000-1200°C, to increase above, when void formation onsets, but **remains less than 1-2% in the whole range**. Deviations with excess Si or C may induce local volume changes. Non-stoichiometric SiC-based fibers are known to undergo substantial volume contraction under exposure to neutrons. Here the **main issues to be addressed concerns swelling in the actual component**: for this reason SiC_f/SiC sandwich composites irradiations have been performed in the BOR60 reactor in 2012-2015, in contact with sodium at 550°C, with doses up 105-120dpa.

The **loss of thermal conductivity of SiC_f/SiC under irradiation by one order of magnitude or more poses the most severe limitations for its application as fuel cladding**. A major role in this is played by the evolution during irradiation of the PyC interphase, through formation of a network of microcracks, due to amorphization and dimensional changes, that induces radial tensile stress at the interface between the PyC and SiC. If the stress exceeds the bond strength between layers, debonding of PyC from SiC occurs. Work should be therefore devoted to understand the mechanisms that drive the microstructural evolution of the PyC with increasing dpa. Ion irradiations can provide only limited information because, contrarily to metals, covalent compounds are severely damaged by electronic excitations, making it especially difficult to extrapolate their behavior under neutron irradiation.

SiC corrosion Issues

Chemical compatibility in reactor environments is highly dependent not only on the thermodynamic stability of SiC in the pure coolants, but also in contact with the possible impurities contained therein. The coolant may contain small amounts of gas impurities such as CO₂, CO, H₂O, H₂, CH₄, O₂, as well as solid particles coming from a variety of sources throughout the reactor system.⁴¹ However, the lack of graphite moderation in the GFR, compared to HTR systems, will significantly reduce the concentration of C-based impurities. Therefore the corrosion of SiC_f/SiC cladding in a GFR will likely occur primarily by oxidation, rather than by any other chemical process. The key oxidising impurities to consider in the He coolant will be oxygen and water vapour, the latter being of greatest importance. **The reaction of SiC with O₂ and moisture at elevated temperatures leads to three typical oxidation features, passive oxidation, active oxidation and bubble formation.** In an environment with sufficient high oxygen concentration, pure SiC is expected to form a SiO₂ passive layer protecting the surface according to the reaction $\text{SiC} + \text{O}_2 \Rightarrow \text{SiO}_2 + \text{C}$. But under oxygen deficient conditions the reaction proceeds as $\text{SiC} + \text{O} \rightarrow \text{SiO} + \text{CO}$, where the SiO is a gas and the reaction proceeds until consumption of the SiC.⁴² Furthermore, in environments containing water vapour at very high temperature (1900K) the formation of SiO and CO vapours at the SiC/SiO₂ interface leads to bubble formation, growth and rupture of the oxide. In active oxidation and bubble formation regions, SiC_f/SiC undergoes therefore significant loss of mass and non-protective SiO₂ layer formation. Thus, the use of SiC composites requires extensive testing to evaluate the impact of non-passivating oxidation and corrosion in relevant environments. Barrier coating to hinder or limit corrosion may have to be envisaged.

SiC Joining

SiC_f/SiC tubes are initially fabricated with one closed end; the open end must then be hermetically sealed after loading the fuel pellets with an end-cap that should withstand the pressure of the fission gasses and the neutron radiation field, while being chemically stable in the coolant environment. **A crucial technology gap is the lack of a reliable, reproducible technique to join and hermetically seal the tubes.** A number of techniques for the joining of ceramic composite materials to themselves or dissimilar materials (e.g. metal components) have been developed. These include spark plasma sintering, laser-based joining, brazing, diffusion bonding, transient eutectic phase routes, glass-ceramic joining, adhesion, pre-ceramic polymer routes and mechanical fastening. Presently there is insufficient information pertaining to the compatibility with coolants and stability under irradiation of the joints and additional studies are needed for the assessment of reliable joining technologies

Standardization of testing procedures for SiC_f/SiC

The qualification and eventual codification of SiC_f/SiC requires a vast effort of pre-normative research, starting from the problem of the standardization of tests and continuing with the need to characterize the behaviour of the material in environment (tightness against fission products, contact with flowing He, irradiation). **Standards on mechanical test for nuclear grade SiC_f/SiC are necessary to produce accurate and reliable data, based on well-defined test methods, detailed specimen preparation, comprehensive reporting requirements, and commonly accepted terminology.** In particular, standard test methods are critically important to achieve predictive capability for failure probability of SiC_f/SiC tubular structures under given loading conditions. Testing is performed at several laboratories to progress on the development of GFR fuel element, so it is mandatory that common and well-recognized sets of testing and evaluation technologies are followed. Lacking these, the work would become inconsistent and difficult to accept by the design community.

A number of standard tests for continuous fiber ceramic composites have already been developed by relevant technical committees of existing National/International Organizations for Standardization: ASTM (C28-07), CEN (TC184-SC1), ISO (TC206/WG4), AFNOR (B43-C). However, not only there are no commonly accepted design methodologies for tubular components made of advanced composites, but there are also no mechanical test standards for any of the properties of tubular geometry ceramic composite components. Also, the temperature range of interest for the clad case is not addressed.

Work on standards for ceramic composites are deliberately separated from that on monolithic ceramics, due their heterogeneous microstructure, unique properties and behaviour, requiring appropriate specimen geometry, experimental procedures and approaches to data analysis. As a consequence, current standards for design purpose are inadequate because they ignore the fundamental issue of whether the specimen they employ is fully representative of the entire composite structure. The failure behaviour of SiC_f/SiC components

⁴¹ Estimation for ALLEGRO from HTR-10 experiments.

⁴² This active oxidation could be an issue for the use of SiC in Na or HLM that contain little oxygen.

having tubular geometries is anticipated to be significantly different from that observed on flat two-dimensional architectures. Developments in this direction are therefore required, with the contribution of qualified laboratories having experience in continuous fiber ceramic composite testing and investigation, possessing the needed equipment.

- **Refractory metallic alloys**

The requirements of safe operations at very high temperature calls for materials with melting temperature above 1500°C. **Refractory metals (V, Nb, Mo, Ta and W) have melting points above 2000° and are thus an obvious choice for high temperature structural applications, as metallic alternative to SiC_f/SiC** and also as competitors to advanced steels for in-core applications. They are potential candidate materials for the commercial deployment of SFR and HLM cooled systems and serve as backup solution for GFR fuel cladding to provide a risk mitigating measure in case of SiC_f/SiC option failure. The high melting temperature implies several beneficial properties such as high elastic modulus, high microstructural stability, high limiting creep rate and low thermal expansion coefficient. **However, these alloys exhibit high affinity for oxygen, hydrogen, nitrogen and carbon, that easily diffuse in the bulk causing hardening and embrittlement during both the production process and the service life.**

The upper temperature limit of applicability is thus mostly determined by oxygen corrosion and interstitial impurities embrittlement. This restricts their high temperature applicability to vacuum or high purity inert atmospheres, unless they are properly protected by suitable coatings to prevent oxidation and mass transfer issues. Canning under inert atmosphere is mandatory for high temperature processing and welding need appropriate processing to avoid contamination.

The low temperature limits as cladding materials are determined by irradiation embrittlement and DBTT shift at or above the operational temperatures. V, Nb, Mo, Ta and W have all DBTT at or above room temperature and the radiation induced defects shift the DBTT to around 400°C in V4Cr4Ti and above 700°C for Mo ZTM.

Another key challenge at present is to re-establish large-scale production capabilities and to recapture former expertise on processing, joining, brazing, cold work in various shapes, irradiation effects, and weldability.

Here only two alloys of this type are further discussed as prominent examples, that were studied to some extent in the past and recently experienced renewed interest for cladding: V and Mo alloys.

V-alloys

Advanced V-alloys with increased operating temperature window are considered in the US and in Japan for fusion reactor blankets. High quality manufacture of V-4Cr-4Ti heats of high purity, with state-of-the art properties required for blanket application, has been demonstrated into a variety of engineering-relevant shapes, including small diameter thin wall tubes for creep studies. These materials have attractive features also for application as fuel cladding materials in fast reactors. Research on V-alloys as clad materials for the previous generation SFR was initiated in Europe during the 1970s but soon discontinued. A programme of fabrication of V-4Cr-4Ti alloy (CEA-J57 grade) cladding tubes for SFR was started in 2008 by CEA and INR,⁴³ but was interrupted in 2015 due to budget constraints. V-alloys, remain since on the list of potential interests for GenVI applications.

The V-4Cr-4Ti alloy has a design temperature window of about 430°C-700°C, for an allowable stress of 100 MPa, that is bounded on the lower side by irradiation embrittlement and on the upper side by thermal creep and recrystallization. Essential condition to make V- alloy potentially suitable for core applications is to enlarge this reference window, by expanding both the low and high operation temperature limits. The issue to be addressed is therefore to find the most appropriate approach (composition, fabrication process, ...) to improve high temperature strength and low temperature ductility of V-alloys, in a way similarly to F/M steels.

Another element of concern is the high vulnerability to oxidizing species and proneness to embrittlement by impurities, unavoidable in fast reactor coolants. Compatibility with fuel or fission products could also arise. These are very significant feasibility issues that impose the development of adequate protective barriers ensuring protection against oxidation and corrosion. Attention should be focused on suitable technologies that avoid contamination, are compliant with operative requirements and nuclear environment, and are applicable to complex surfaces, including tubes and their internals.

Mo-alloys

Molybdenum is probably the most versatile among the refractory metals: easily machined with conventional tools, it has high strength and rigidity at elevated temperatures (60–80 MPa at 1500°C), excellent thermal conductivity (138 W m⁻¹K⁻¹), good resistance to swelling under irradiation and acceptable neutron capture cross

⁴³ FP/ MATTER Report D11.2.

section. Mo-alloys have been mainly considered in high temperature fast reactors for space applications. Due to its neutron capture cross section, Mo and its alloys have been excluded from the list of candidate materials for the high temperature evolution of the SFR. However, the lower neutron absorption cross section of HLM and He cooled systems makes their use viable in LFR's and GFR's cores. Main drawbacks are the limited ductility and irradiation induced embrittlement with DBTT shift, depending on dose and irradiation temperature, up to 800°C for unalloyed Mo. Impurities influence the mechanical properties, particularly through grain boundary weakening, but improvements in ductility are achievable through grain refinement, impurity control, and alloying, so these would be issues to address. Grain size reduction and the uniform dispersion of fine La_2O_3 oxide particles at GBs and in the interior of the grains in ODS Mo alloys have been found particularly effective in increasing strength and ductility. Molybdenum-lanthanum ODS show DBTT at -100°C in the unirradiated state and outstanding reduction in the DBTT shift after irradiation even at low temperatures: improvements largely due to the grain size reduction and the presence of the oxide particles that strengthen the grain boundaries and offer additional sinks for point defects. Creep performance is however a potential issue to be evaluated for applications as cladding material.

The corrosion behaviour in oxidising environment of Mo-based alloys is an issue, due to the formation of MoO_3 , which volatilizes easily above 400°C leading to rapid oxidation, potentially catastrophic at higher temperatures. However, Mo has good corrosion resistance in several liquid metals including Bi, Li, K, Pb and Na. It has high corrosion resistance and poor solubility in liquid Pb and LME up to 800°C. This limited corrosion in HLM gives Mo alloys potential as cladding for the HLM cooled systems. Yet, depending on the mechanisms leading to the observed corrosion resistance, their application as cladding may require the deposition of suitable diffusion barrier to prevent mass transfer issues and disruptive oxidation. The feasibility of Mo-alloys fuel cladding for HLM cooled reactors should be thus demonstrated via: (a) Characterization of the corrosion in HLM varying [O] and temperature; (b) characterization of the creep performance of the ODS-Mo alloys; (c) mechanical testing to investigate LME issues; (d) development of welding procedures; (e) development of barrier coatings to prevent gathering of the oxygen contained in the coolant; (f) irradiation at dpa doses relevant for GenIV, to evaluate swelling and embrittlement issues.

- **Modified surface layers for protection against coolant attack**

The adoption of corrosion-resistant materials and the mitigation strategies set in place to limit the effects of aggressive media may not be sufficient or applicable, thereby calling for protection via coatings or other surface treatments. The underlying idea is that **protective surface layers should not alter the mechanical properties of codified materials with known properties, but would impart the desired corrosion (and/or wear) resistance**. The previous sections show that this need is common to (a) steels in HLM operating at temperatures above 400-450°C; (b) SiC_r/SiC in oxygen deficient high temperature environments; (c) Refractory alloys.

There exist four main criteria for the development of an optimized surface layer (SL):

- 1) The SL should form thermodynamically stable protective phases by reaction with the environment.
- 2) These phases should be slowly growing in order to keep SL reservoir depletion rates low.
- 3) Interdiffusion between layer and substrate should occur as slowly as possible, suggesting the introduction of an interdiffusion barrier or substrate where the diffusion rate of the SL species is low.
- 4) The values of the coefficients of thermal expansion of SL and substrate should be very similar, to minimize cooling and reheating stresses during temperature oscillations.
- 5) If deposited, the SL deposition processes should be carried out at low temperature to avoid the degradation of the steel performance, in general and especially under neutron irradiation.

The last constraint rules out the adoption of diffusion coating techniques for the core components that imply exposure to temperatures > 800°C (in steels) for times of the order of hours.

Most work on protective SL for core applications in ESNII prototypes involves two systems:

- alumina-forming metallic layers produced by the GESA process;
- ceramic Al_2O_3 barrier coatings produced by pulsed laser deposition (PLD).

Other processes are of course open and possible but these two are discussed here as prominent examples.

GESA process

This process consists in the deposition of an Al-containing metallic layer on the steel surface, subsequently melting the layer, together with a few μm from the substrate, using intense pulsed electron beams.⁴⁴ During exposure to an oxygen-containing medium, if the concentration is sufficient, a thin, continuous, slowly growing, adherent and stable alumina layer is formed at the surface, that protects the components from corrosion attack.

No sign of dissolution attacks or exfoliation was observed during corrosion tests, performed at temperatures between 400-650°C, for up to 10000 h. During creep-to-rupture, cycling fatigue, bending, pressurized tubes and erosion tests, the alumina layer protected the steel components with modified SL against the negative influence exerted by the HLM on the mechanical properties. The modified SL have shown also promising irradiation tolerance.

However, for qualification of this material concept, R&D is needed on: (a) optimization of the SL composition; (b) optimization of the deposition method; (c) exposure to HLM in normal and off-normal conditions; (d) ageing in HLM; (e) welding of steel components with modified SL (e.g. end caps); (f) irradiation tests at relevant doses in presence of HLM.

Pulsed lased deposition (PLD)

In this case an $\text{Al}_2\text{O}_3/\text{FeCrAlY}$ system that includes barrier coating and buffer layer is deposited using a pulsed laser. The buffer layer accommodates the stresses due to the differences in the thermomechanical behaviour. Coatings deposited on F/M steels have shown after testing no need of buffer layer. PLD is a relatively simple technology that allows thin films to be grown on a wide range of materials. The deposition conditions can be adjusted to obtain different coating microstructures, varying from fully dense and compact to columnar and porous. Fully dense and compact Al_2O_3 coating grown by PLD on various steels have been obtained by tailoring the deposition process to attain an advanced nanocomposite which consists of a homogeneous dispersion of ultra-fine nanocrystalline domains, with size in the range 6 ± 4 nm in an amorphous alumina matrix. The material gives an ensemble of metal-like mechanical properties ($E=195\pm 9$ GPa, $\nu=0,29\pm 0,02$) enhanced plastic behaviour, relatively high hardness ($H=10$ GPa), full compactness and strong adhesion. Al_2O_3 barriers have been tested up to 600°C in stagnant HLMs with outstanding results and their performance under heavy ion irradiation investigated up to 150 dpa at 600 °C, the energy of the ions having been chosen to obtain the maximum displacement damage at the interface with the substrate. The irradiation did not induce any loss of adhesion or delamination effects.

Complete characterization with pre-normative purpose will require evaluation of: (a) performance under neutron irradiation; (b) fracture toughness by micro-indentation; (c) resistance to high speed erosion by HLM.

Detonation gun spraying

Detonation gun spraying is a thermal spray process to deposit overlay thick, hard coatings on steels, with an extremely good adhesive strength, low porosity and only affected by compressive residual stresses, for parts subjected to wear and erosion/corrosion. The combustion of a gas mixture, ignited by a spark plug, in the detonation chamber generates high pressure shock waves, which then propagate through the gas stream. The hot gases produced travel through the barrel at a high speed and high temperature, carrying the material to be deposited in a plastic state with velocity of 1200 m/s. The particle impacts on the surface form a splat and the coating results from several splats. The temperature and speed of the particles lead to dense and adherent layers. Due to the size of the splats and the residual stresses generated during the cooling down, the minimum applicable protective layer is around 10 μm . Although the powders reach temperatures up to 4000°C, the deposition does not affect the substrate temperature, which remains cold. The residual stresses can be reduced by interposing a buffer layer on the substrate. Work is needed, here too, to optimise the process and to evaluate the performance of the coating under service conditions. Because of the step variation of the mechanical properties between steel substrate and ceramic coating, cyclic loads are expected to be especially damaging.

Qualification of coated materials

The qualification of coated materials is especially delicate because, having in mind the licensing of the component, it will be necessary to prove that in case of (local) failure of the SL protection no major safety-threatening consequences will ensue. Essentially this imply foreseeing not only tests aimed at verifying the stability of the protective SL and standard qualification procedures (see section 3.1.1, page 39), but also to include in any case the qualification of the substrate material, in order to know which corrosion-rate or effects

⁴⁴ Patent: EP1896627; cladding tubes made of ferritic/martensitic or austenitic steel for nuclear fuel elements/fuels and method for subsequently treating a FeCrAl protective layer thereon that is suited for high temperatures.

in general should be expected in case of SL failure, as well as, crucially, *worst case condition* testing, according to criteria that will necessarily depend case by case.

- **Prospective materials**

None of the materials solutions listed in the previous sections can be truly called new or innovative. These are all materials that have been long considered, but that still require significant work in terms of composition, manufacturing processes and property screening before a final codifiable material emerges and reaches sufficiently high TRL. Here two truly new classes of materials with promising features, but for the moment never used in any technological application, are discussed, namely high entropy alloys (HEA) and MAX phases.

High-entropy alloys

High entropy alloys (HEA) are a fundamentally new metallic material concept proposed in recent years.⁴⁵ Elements are combined in roughly equimolar concentrations so that the high entropy of mixing stabilizes simple solid-solution phases with relatively simple crystal structures, hindering the formation of conventionally expected complex intermetallics. Thus, while these alloys may be compositionally complex, they are microstructurally simple. HEAs exhibit high strength due to their compositional complexity (solute strengthening), being thus considered promising for high temperature applications. HEAs containing passivating elements, such as Cr, Al, etc., have shown equivalent or superior resistance to corrosion compared with conventional alloys in aggressive environment.⁴⁶ These good mechanical and corrosion properties make HEAs attractive wherever extreme service environments exist, such as in nuclear. Relatively little is known, however, about their stability under neutron or ion irradiation. The metastability associated with the energy stored in the lattice distortions, could ultimately lead to the evolution toward the amorphous state under irradiation, or induce phenomena of segregation and precipitation against thermodynamic forces. Although existing results indicate that they can be excellent irradiation-resistant materials.

The concept of a crystalline phase stabilized by the entropic contribution of the free energy is extremely attractive and offers the possibility to tailor via a suitable alloy design the desired thermo-mechanical, corrosion and radiation resistance properties, by modifying the composition without *a priori* incurring in precipitation and segregation issues. The field offers therefore wide opportunities to explore, discover, and develop new classes of alloys for structural and functional applications. Alloying element combinations previously perceived as questionable, due to microstructure instability, may now become a possibility, suggesting completely new families of light metal alloys, high strength metals, and high-temperature metals. Further research could be devoted to explore their applicability to the GenIV systems. Beyond the necessary efforts required to develop a totally new class of alloys, that involves the synthesis, mechanical testing, corrosion testing, irradiations etc., it is important to emphasise that in this case alloy design cannot currently be supported by CALPHAD type models: databases need to be developed, opening a wide field also for modelling as guidance for an accelerated development (section 3.1.2).

MAX phases

The MAX phases are layered solids with hybrid metallic-ceramic behaviour and properties that depend on stoichiometry, given by the general formula M_n+1AX_n , where M is an early transition metal, A is an A-group element (Al-S, Ga-Se, Cd-Sb, Tl-Bi), and X is C or N, while n is typically 1, 2 or 3. They are versatile materials, whose properties (deformability, thermal stability, oxidation resistance, etc.) can be tailored by forming solid solutions on the M, A and X sites, that often exhibit better properties than the 'parent'. The machinability of the MAX phases is similar to that of graphite,^{Erreur ! Signet non défini.} which makes them suitable materials for the production of geometrically complex components. Finally, these materials are characterized by unusually high –for ceramics– damage tolerance (e.g. K_{IC} values of up to $18 \text{ MPa} \cdot \text{m}^{1/2}$ were reported for textured Nb_4AlC_3), due to various toughening mechanisms. In terms of response to irradiation MAX phases seem to have a remarkable capacity for self-annihilation of neutron-induced defects at elevated temperatures.

Because of their excellent compatibility with HLM they are promising core materials for HLM-cooled GenIV systems. For example, exposing Ti_3SiC_2 to HLM with [O] in the 10^{-8} - 10^{-6} mass% range, between 550-750°C range and up to 4000 h showed the formation of a thin oxide (TiO_2) scale and no liquid metal attack. Moreover,

⁴⁵ Yeh JW, Chen SK, Lin SJ, Gan JY, Chin TS, Shun TT, et al. Adv Eng Mater 6 (2004) 299-303; Huang PK, Yeh JW, Shun TT, Chen SK. Adv Eng Mater 6(1-2) (2004) 74-8; Yeh JW. Ann Chim Sci Mat 31(6) (2006) 633-648; Zhang Y, Zhou YJ, Lin JP, Chen GL, Liaw PK. Adv Eng Mater 10 (6) (2008) 534-538.

⁴⁶ Yi-xuan LIU, Cong-qian CHENG, Jian-lu SHANG, Rui WANG, Peng LI, Jie ZHAO Transactions of Nonferrous Metals Society of China Volume 25 (4) (2015) 1341-1351.

screening mechanical tests on selected MAX phases in oxygen-poor LBE at 350°C showed no mechanical property degradation.

As with every innovative nuclear material, the MAX phases need to be optimised for the envisaged application. Optimisation involves:

Selection of appropriate composition followed by microstructural tailoring, playing with the possibility of forming solid solutions M, A and X sites, so as to meet the property requirements of the targeted end application. The appropriate composition can also be determined in order to limit the end-of-life component activation.

Phase purity: it is often challenging to produce MAX phase materials without ‘parasitic phases’, such as binary carbides and intermetallics. Phase purity can be seriously improved by making solid solutions or by the addition of critical dopants, but the right approach to making phase-pure MAX phases is chemistry-specific and labour-intensive. Phase purity is likely to affect both the radiation tolerance of the MAX phase materials (possible cracking due to differential swelling) and their corrosion resistance.

Collection of statistically-relevant experimental data: the design and optimisation of MAX phases for selected Gen-IV applications involves application-driven material processing, mechanical testing in both inert and liquid metal media, corrosion/erosion testing and irradiation. The collection of statistically-relevant data could be accelerated with appropriate design/production/performance assessment strategy.

• **Summary on structural materials issues**

Because of the large number of structural materials classes and types, it is convenient to provide a summary of the main issues connected with pre-normative research, modelling and advanced materials solutions. This is done in **Table 7**.

Table 7: Summary of structural materials and relevant issues.

| Type of related issues | | Pre-normative research | Modelling | Advanced materials' solutions |
|---|--|---|---|--|
| Austenitic steels | 316L(N) (prototype irreplaceable components) | Thermal ageing, thermal creep, compatibility with heavy liquid metals (HLM)/gas: increased database (including welds), accelerated testing, models describing micro/macro evolution → refinement of existing, or elaboration of new, design rules | | Improve compatibility with coolants, apply high temperature protective barriers |
| | 15-15Ti (cladding) | Irradiation creep and swelling, thermal creep, compatibility with coolants & fuel: increased database, models describing micro/macro evolution → refinement of existing, or elaboration of new, design rules | | Increase swelling resistance and compatibility with coolants |
| | Alumina forming austenitic (AFA) steels | Exposure needed for screening between candidates | Thermodynamic models for composition optimisation, microstructure evolution models | Addition of Al increases compatibility with coolants (protective alumina layer), but causes embrittlement at low T, although improves high T creep strength (NiAl precipitates): compromise searched |
| Ni-based alloys (heat exchangers, valves, coaxial pipes) | Alloy 800 (high Ni austenitic steel) | Design properties are available for Alloy 800H from existing Codes and Standards; Need to validate the properties of thin-section under extreme conditions due to strength reduction. | To identify mitigation strategies, models of radiation-induced microstructure evolution in connection with predictions of embrittlement and swelling (He production). | Improve compatibility with alternative coolants and high temperatures; increase strength properties |
| | Actual Ni-based alloys (eg Inconel 617, Haynes 230, ...) | Exposure to high temperature in environment needed for screening between candidates and then for qualification. | Creep and creep-fatigue engineering models in support of design correlations and rules. | Compatibility with coolant at high temperatures; manufacturing and joining |

| | | | | |
|---|--|---|---|--|
| | | Not suitable for irradiation field environments due to swelling and embrittlement. | | |
| Ferritic / Martensitic (F/M) steels (cladding and core) | 9-14 %Cr | Low temperature irradiation embrittlement, irradiation creep, thermal ageing/creep, creep-fatigue (cyclic operation softening) compatibility with coolants, liquid metal embrittlement: increase database (including welds), models → define design rules and develop models in support | | Need solution to minimize embrittlement, improve creep resistance (e.g. by thermomechanical treatment) and improve compatibility with coolants |
| | Oxide dispersion strengthened (ODS) | Exposure needed for screening between candidates, suitable treatments for recrystallization to eliminate anisotropy after powder metallurgy production of bars and tubes by extrusion. | Oxide formation/stability, microstructure evolution, modes of deformation | ODS steels (tubes) have better creep resistance, but manufacturing and joining are issues (optimization needed); toughness and compatibility are also issues |
| | FeCrAl alloys (also ODS) | | Thermodynamic models for composition optimisation, microstructure evolution models | Addition of Al increases compatibility with coolants (protective alumina layer), but worsens mechanical behaviour: compromise searched |
| Refractory metallic alloys (cladding and core) | Molybdenum alloys (including ODS) | Exposure needed for screening between candidates. Irradiation creep and swelling, thermal creep, compatibility with coolants & fuel: increase database, models describing micro/macro evolution → refinement of existing, or elaboration of new, design rules, supported by models | | Prospective materials, mainly for cladding, studied also in the past, with problems of manufacturing, compatibility with coolant and mechanical behaviour |
| | Vanadium alloys | | | |
| | High Entropy Alloys | Prospective metallic materials with potentially excellent mechanical properties, coolant & radiation resistance, need extensive investigation for screening, including understanding of origin of properties through modelling, before applications are identified | | |
| Ceramics (cladding and coating) | SiC/SiC (also C/C) composites (cladding) | Mechanical test standardization, radiation resistance (thermal conductivity, hermeticity, swelling, ...) and corrosion resistance → define design rules | Microstructure evolution models under irradiation, finite element models for composite architectures, X-ray tomography techniques | Liners to guarantee hermeticity of cladding, or other techniques to guarantee hermeticity. Limit thermal conductivity degradation under irradiation. |
| | Graphite | Irradiation effects on oxidation resistant graphite, irradiation creep, Codes & Standards development | Dependence of properties on porosity, graphite structure dynamics (stress states) | SiC/Graphite “composites” |
| | Non-metallic core support structures (<i>ad hoc</i> ceramics) | Screening of candidates. Test standardization (mechanical & thermophysical properties) | Microstructure evolution models under irradiation | Protection against oxidation |
| | Al ₂ O ₃ coatings | Applied with different techniques on different substrates to protect against coolant attack and temperature: exposure for screening and qualification | | |
| | Max phases | Prospective ceramic materials with excellent mechanical properties (for ceramics), coolant and radiation resistant, need extensive investigation for screening, including understanding of origin of properties, before applications are identified. Usable as coatings. | | |
| | | | | |

3.2 Fuel materials

3.2.1 Fuel materials qualification and design rules

This section addresses the development of materials data and simulation tools in support of the design and licencing of fuels for ESNII prototypes.

Although the processes occurring in pile in nuclear fuels are interconnected, the fuel material properties and processes governing the behaviour of nuclear fuel under irradiation can be grouped in five main categories of issues, as follows.

- ***Margin to fuel melting***

The maximal temperature of the fuel at the centre of the pin is a key safety parameter since design rules indicate absence of melting as a criterion to be respected at all time. The margin to melting is defined as the difference between the melting point (solidus temperature) of the fuel in its actual state and the centreline temperature of the fuel. This criterion is all the more important for ESNII reactors, which will operate at high temperature.

Therefore, the melting point and the actual maximal fuel temperature in pile, which are both largely dependent on the fuel composition, chemistry and burn-up (BU), must be known with great accuracy. This necessitates a comprehensive knowledge of the system phase diagrams and of the evolution of the thermal conductivity and specific heat as a function of the fuel parameters.

- ***Atom transport properties and consequence on the microstructure***

The atom transport properties are at the origin of several important phenomena taking place during irradiation, for instance the redistribution of Pu in the fuel, which strongly affects the heat and fission product distribution, and the oxygen diffusion, which governs the local oxide/metal (O/M) ratio for oxide fuels. Furthermore, solid-state diffusion is involved in the drastic restructuring of fast reactors fuels taking place in the early stages of irradiation, alongside pore migration, a consequence of the very high temperature gradient across the fuel pellet radius.

These atom transport properties are strongly influenced by the atomic-scale defects created by energetic particles produced through fissions and radioactive decay, as well as intense ionization.

Reliable data and in depth understanding of the key controlling phenomena are essential for accurate and reliable modelling and assessment of the consequences of the relocation effects taking place in FR MOX in fuel performance code.

- ***Fission Product and Helium behaviour***

The cladding constitutes the second retention barrier against radioactive fission product release. A design criterion is the absence or minimisation of mechanical interaction between the fuel and the cladding. To calculate the mechanical load exerted on the cladding, it is necessary to know, *inter alia*, the quantities of fission gases and helium released to the pin plenum and the pressure generated. The fuel swelling, the quantity of gas retained in the fuel and its location (in solution in the lattice, at grain boundaries, or in bubbles, intra or intergranular), must also be known.

In addition, the increasing inventory of non-gaseous fission products in the fuel with burnup and time has a substantial effect on the fuel chemistry, including for instance the oxygen or carbon potential of the fuel.

- ***Mechanical properties***

The load on the cladding depends the mechanical evolution of the fuel in the reactor since it governs the fuel-cladding mechanical interaction (FCMI), which becomes significant during permanent low power operation or in case of a power increase after a long low-power operation. The mechanical properties also come into play in gap closure, which has a supreme influence on the temperature of the fuel and the corrosion of the cladding, but whose mechanisms are largely unknown.

- ***Compatibility between fuel, cladding and coolant***

Among the fission products created under irradiation, volatile elements such as caesium, tellurium, iodine, as well as molybdenum, migrate from the centre to the periphery of the fuel pellet to form, in the case of oxide fuels, the “oxide-clad joint” (OCJ) layer. This layer constitutes a second potential interaction risk between the fuel and the cladding. The fuel-cladding chemical interaction (FCCI) results eventually in the corrosion of the

cladding, the “oxide-clad reaction” (OCR). This corrosion is another major factor limiting the integrity, and therefore the lifetime, of the fuel pin, eventually impinging safety and economics of operation.

In addition, in case of a breach in the cladding (pin failure), the primary coolant can enter the fuel element and come in contact with the fuel, which can lead to a chemical reaction, in particular between MOX and liquid metal coolants, such as sodium lead or lead-bismuth. This reaction could lead to a further degradation of the fuel pin and potentially to the dissemination of fuel in the coolant. Thermodynamic and kinetic aspects of these reactions are still unknown, especially at high burn-up.

The properties and mechanisms relative to these issues need to be known with accuracy for the improvement of fuel performance codes, which play an essential role in the qualification of innovative nuclear fuels.

This qualification focuses on the measurement of these properties. It relies on integral irradiation testing (full length pins and assemblies) and involves the examination of irradiated fuels with a known irradiation history representative of the conditions of ESNII reactors, as well as corresponding measurements on fresh fuels for reference.

Irradiation testing is therefore necessary either in material testing reactors, or in the limited number of fast neutron reactors available today. Then, traditional post-irradiation examinations (PIE), including the measurement of fuel pin dimension changes, fission gas release (FGR), volatile and non-volatile material migration, microstructure evolution, etc., must be performed in specific hotlabs enabling the handling and characterization of irradiated fuels. The advances made in measurement methods during the last decades permit greater information and knowledge harvesting than in the past, with thermal conductivity, fission gas retention, and fuel vaporization behaviour being now readily attainable. New PIEs should also be performed on “treasure” materials from past irradiation that have been held in storage to reap their full potential.

As indicated in section 2.2, in the short term, all ESNII reactor first loads will rely on MOX fuels. The qualification and understanding of these fuels is thus the focus of this section.

Most of the models and properties used today in the macroscopic scale simulation of nuclear fuels are derived from the 1990 Fast reactor Data manual, which gathered the recommendations on properties of (U,Pu)O₂ fuel made by a group of European experts⁴⁷.

The FP7 ESNII+ project⁴⁸ (2013-2017) has started the update of this property catalogue by continuing the characterization of irradiated fuels and reviewing the results obtained since the catalogue was published to assess the impact of the new results on the recommendations. Thermal properties (thermal conductivity, specific heat, melting temperature and emissivity), structural and mechanical properties (lattice parameter, thermal expansion, elastic constant, brittle-to-ductile transition temperature, yield and ultimate stresses, thermal creep) and atomic transport properties (diffusion and migration of oxygen, uranium, plutonium or minor actinides, fission gases, as well as of fission gases pores, fission gas bubbles) were particularly considered.

The data to be measured to fill the gaps in knowledge that are consistent among others with the final recommendations of the ESNII+ project, as well as the materials concerned and the techniques that can be used are listed in Table 8.

Table 8: Main issues concerning fuel materials qualification and design rules

| Main issue | Breakdown in sub-issues | Materials concerned | Techniques/ Methods |
|-------------------------------------|---------------------------------|--|--|
| Evolution of margin to fuel melting | Establishment of phase diagrams | Irradiated oxide fuels and fuel pins: MOX (U,Pu)O ₂ UO ₂ | Determination as a function of O/M, Pu, Am content, BU of <ul style="list-style-type: none"> ▪ Melting points ▪ Heat capacity ▪ Vapour pressure Techniques: laser heating, calorimetry, Knudsen cell mass spectrometry (KEMS) |
| | Evolution of thermal properties | (U,Pu,Am)O ₂ (U,Am)O ₂ Fresh oxide fuels as reference | Measurement as a function of temperature, composition (O/M, Pu/M, BU) of <ul style="list-style-type: none"> ▪ Thermal conductivity ▪ Thermal diffusivity ▪ Emissivity Techniques: laser flash analysis |

⁴⁷ Fast Reactor Data Manual, Technical report NT CEA DRN/DEC/SPU/LPCA 2 (1990)

⁴⁸ <http://www.snetp.eu/esnii/>

| | | | |
|---|---|--|--|
| Atom transport and microstructural evolution | Atom transport | | <p>Measurement as a function of temperature and initial composition (O/M, Pu/M, BU) of</p> <ul style="list-style-type: none"> ▪ Lattice parameter ▪ Diffusion of U, Pu, Am ▪ Diffusion of O ▪ Oxygen potential <p>Techniques: SEM-WDX, EPMA, SIMS, coulometric titration, XRD</p> |
| | Microstructural Evolution | | <p>Measurement as a function of temperature (radial position), initial Pu/M, O/M ratios and porosity and BU of</p> <ul style="list-style-type: none"> ▪ Density ▪ Beginning of life restructuring ▪ Centre void formation ▪ Pu, Am and O/M homogeneity and content ▪ Diffusion/migration of pores ▪ Grain size and their distribution <p>Techniques: Optical microscopy, SEM/WDX-EDX, TEM, EPMA, SIMS, XRD</p> |
| Fission Products and Helium | Gas (fission gas and He) behaviour | | <p>Measure of gas release in irradiated fuel as a function of BU and irradiation history (normal, off-normal and severe accident conditions)</p> <p>Techniques: gas puncturing, KEMS, thermal desorption spectroscopy</p> |
| | Non-gaseous FP transport | | <p>Measure of fission products transport, release and compounds formed as a function of BU and history (in normal, off-normal and SA conditions)</p> <p>Techniques: Gamma scanning of fuel pins, KEMS, optical microscopy, SEM/WDX-EDX, EPMA, TEM, XRD, ICP-MS.</p> |
| Mechanical properties and mechanical interactions between fuel and cladding | Evolution of mechanical properties and “state” of fuel pellets Fragmentation, cracking | | <p>Measurement as a function of temperature, Pu/M and O/M ratio, microstructure and BU of:</p> <ul style="list-style-type: none"> ▪ Thermal expansion ▪ Elastic constants ▪ Ultimate stress ▪ Yield stress ▪ Brittle to ductile transition temperature ▪ Hardness ▪ Creep rates <p>Techniques: XRD, dilatometry, acoustic methods, micro-nano indentation, eventually miniaturised mechanical testing methods</p> |
| | Fuel Cladding Mechanical Interactions | | <p>Measurement as a function of temperature, Pu/M and O/M ratio, microstructure and BU of:</p> <ul style="list-style-type: none"> ▪ Pellet geometry ▪ Inner pin geometry ▪ Swelling ▪ Pellet density ▪ Determination of remaining gap width between pellet and cladding <p>Techniques: Optical inspection, XRD, dilatometry, pin profilometry, immersion method, optical microscopy, SEM, neutron radiography</p> |
| Chemical interactions between fuel, cladding and coolant | Chemical interactions between fuel and cladding, internal | Irradiated and fresh fuel pins, Breached pins, | <p>Determination of</p> <ul style="list-style-type: none"> ▪ Width of corrosion layer ▪ Composition of corrosion layer <p>Techniques: Gamma scanning, SEM/WDX-EDX, optical microscopy</p> |

| | | | |
|--|---|--|---|
| | cladding corrosion | Sodium, lead and lead-bismuth coolants | |
| | Chemical interactions between fuel, cladding and coolant in case of severe accident | | Measurement (as a function of BU if possible) of: <ul style="list-style-type: none"> ▪ Size of breach ▪ Amount of fuel / fissile matter loss ▪ Phases produced Techniques: Optical inspection, Gamma scanning, SEM/WDX-EDX, optical microscopy |

3.2.2 Advanced fuel materials modelling and characterization

As stressed in Section 2.3, a combination of advanced modelling and separate effect experiments including detailed materials characterization is used in complement to technological research. On fuels, this involves basic research investigations on the five issues described in the previous section.

The aim is to use the approach described in Section 2.3.2 and combine the results to those of the pre-normative research to unveil the missing relevant data and elementary mechanisms underpinning the fuel behaviour and to extend the reliability regime of traditionally deduced empirical laws governing various aspects of nuclear fuel under irradiation.

This advanced modelling and characterization approach has started later on fuel compounds than for structural materials. Nuclear fuels are usually insulators or semi-conductors and have therefore specific thermo-mechanical and transport properties. The defects generated by irradiation are also significantly more complex than in metals, for instance because of their electric charge. The modelling of nuclear fuels is also particularly challenging because of the complex behaviour of $5f$ electrons in actinide compounds.

This type of study progressed significantly on uranium fuels, especially UO_2 and to a lesser extent uranium carbide and uranium nitride, in the F-BRIDGE FP7 project (2008-2012).⁴⁹ Basic data such as diffusion coefficients of oxygen, cation and He, melting temperatures and consistent thermodynamic data were determined using new measurements in well controlled conditions (oxygen potential for fuels for instance) and samples (chemical composition, impurity rates, density,...), as well as state-of-the-art modelling from the atomic to the grain scale. Emphasis was put on improving the reliability of the data obtained. Results have in particular shown that it is now feasible to use electronic structure and empirical potential calculations to obtain precise data on fuels to feed higher scale models and help interpret experiments on fuels.

In addition, the links between the scales and between modelling and experiments were strengthened. On the one hand, the multiscale modelling exercise on transport properties in uranium dioxide stressed the links built between the atomic and mesoscopic modelling on fuels by synthesising the data needed as input in mesoscale modelling that were calculated at the atomic scale. On the other hand, F-BRIDGE demonstrated a first success in updating existing fuel performance codes using advanced material properties and models obtained from basic research and the multiscale modelling approach.

Finally, data request lists gathering key technological issues, pending scientific questions and corresponding basic research investigations to be carried out were built from the interaction between F-BRIDGE participants and industry representative members of the user group. These lists were at the origin of many investigations started among other in the Joint Programme of Nuclear Materials.

The ICME approach on nuclear fuels has also been reviewed by the Expert Group on the Multiscale Modelling on Fuels of the Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems (WPMM)⁵⁰ established under the auspices of the OECD NEA Nuclear Science Committee. A first state-of-the-art report on the multi-scale modelling of nuclear fuels synthesising the modelling approaches from the atomic to the macroscopic scale devoted to nuclear fuels in support of current fuel optimisation programmes and innovative fuel was released in 2015⁵¹. This report also includes critical analyses of the mid- and long-term challenges for the future, i.e. approximations, methods, scales, key experimental data, characterisation techniques missing or to be strengthened.

The approach must now be extended to the various types of FR fuel materials: MOX and minor actinide bearing oxide fuels, but also actinide carbides and nitrides. It also requires the use of a large number of facilities: hot labs, materials research reactors, large facilities accepting radioactive materials, such as ion accelerators or

⁴⁹ <http://www.f-bridge.eu/>

⁵⁰ https://www.oecd-nea.org/science/wpmm/expert_groups/m2f.html

⁵¹ <https://www.oecd-nea.org/science/docs/2015/nsc-r2015-5.pdf>

synchrotrons for the experimental characterization, as well as supercomputer centres for the modelling. The new data and models obtained will then be implemented in fuel performance codes to enhance their reliability in normal and off-normal situations.

The modelling and characterizations to be done on the issues, as well as the methods and techniques to be employed, are listed in Table 9.

Table 9: Main issues concerning modelling and characterization

| Main issue | Breakdown in sub-issues | Materials concerned | Techniques/ Methods |
|--|---|---|--|
| Temperature: Margin to fuel melting | Phase diagrams: phases as a function of composition | Fresh, self-ion or neutron irradiated MOX (U,Pu)O ₂ | <ul style="list-style-type: none"> ▪ Laser heating measurement of melting temperature of virgin samples with various stoichiometries ▪ Calorimetry on virgin samples with various non-stoichiometries ▪ Atomic scale calculations of melting point and calorific capacity vs non-stoichiometry ▪ Calphad modelling |
| | Evolution of melting temperature with BU | | <ul style="list-style-type: none"> ▪ Laser heating measurement of melting temperature of simfuels, self-irradiated and neutron-irradiated samples (short times, T and flux controlled) with various compositions and non-stoichiometries |
| | Evolution of thermal conductivity with BU | | <ul style="list-style-type: none"> ▪ Calorimetry on simfuels, self-irradiated and irradiated samples with various compositions ▪ Atomic scale calculations of thermal conductivity of fuels vs non-stoichiometry and composition |
| Irradiation defects and atom transport | Irradiation defects: point and extended defects | UO ₂ (U,Pu,Am)O ₂ (U,Am)O ₂ UC (U,Pu)C UN (U,Pu)N (Pu,Zr)N SIMFUELS (fresh fuels including FP) | <ul style="list-style-type: none"> ▪ Calorimetry of virgin, ion-irradiated, self-irradiated and neutron-irradiated samples (short times, T and flux controlled) for various non-stoichiometries ▪ Determination of type of defects created using positron annihilation and Raman spectroscopies, MAS-NMR, XAS, electrical conductivity measurement ▪ Atomic scale modelling of point defects to determine most stable configurations as a function of non-stoichiometry ▪ Atomic scale modelling of extended defects to determine most stable configurations and mechanisms of formation and growth ▪ Atomic scale modelling of displacement cascades |
| | Pu relocation and O/M variation | | <ul style="list-style-type: none"> ▪ Measurement of thermal, irradiation-induced and irradiation-enhanced diffusion coefficients for cations as a function of stoichiometry and composition using ion and neutron irradiated samples, radioactive tracers, electrical conductivity measurements, SIMS ▪ Diffusion couple experiments, including with high temperature gradient ▪ Atomic scale modelling of thermal, irradiation-induced and irradiation-enhanced diffusion coefficients for cations vs stoichiometry and composition ▪ Mesoscale modelling of diffusion to determine cation concentrations in grain/pellet (KMC, rate theory, phase field...) |
| | Microstructural evolution | | <ul style="list-style-type: none"> ▪ Modelling of grain growth, fragmentation ▪ Mesoscale modelling of pore evolution, central void formation ▪ Diffusion experiments with high temperature gradient (laser heating) |
| Fission Product and | Gas behaviour | | <ul style="list-style-type: none"> ▪ Measurement of thermal, irradiation-induced and irradiation-enhanced diffusion coefficients for fission gases and helium as a function of stoichiometry and composition (ion implantation, short and controlled |

| | | | |
|--|--------------------------------------|---------------------------------------|---|
| Helium behaviour | | Fresh, self-ion or neutron irradiated | <p>neutron irradiation, TDS, SIMS, mass spectrometer coupled to Knudsen cell)</p> <ul style="list-style-type: none"> Atomic scale modelling of gas incorporation, as well as thermal, irradiation-induced and irradiation-enhanced diffusion vs stoichiometry and composition Mesoscale modelling of gas concentrations in grain/pellet (in solution, bubbles, grain boundaries) and release |
| | Non-gaseous FP transport | | <ul style="list-style-type: none"> Measurement of thermal, irradiation-induced and irradiation-enhanced diffusion coefficients for fission products as a function of stoichiometry and composition (SIMS, mass spectrometer coupled to Knudsen cell) Atomic scale modelling of FP incorporation, as well as thermal, irradiation-induced and irradiation-enhanced diffusion vs stoichiometry and composition |
| | FP compounds and JOG | | <p>MOX (U,Pu)O₂ UO₂ (U,Pu,Am)O₂ (U,Am)O₂ UC (U,Pu)C UN (U,Pu)N (Pu,Zr)N SIMFUELS (fresh fuels including FP)</p> <ul style="list-style-type: none"> Electronic structure (and empirical potential?) calculations of stability and thermodynamic data on fission product compounds (Cs, I, Te, Mo) Electronic structure (and empirical potential?) calculations of stability and thermodynamic data and compounds between FP and fuel elements (uranates, plutonates...) Synthesis of selected compounds and measurement of thermodynamic data (melting temperature, thermal conductivity...) Development of thermodynamic database describing Gibbs energy functions of the phases of the (Cs-I-Te-Mo-O) as a function of temperature and composition Extension of this database to (Cs-I-Te-Mo-U-Pu-O) system Using models developed above, thermodynamic equilibria calculations to predict the phase formation with respect to oxygen potential and temperature |
| Mechanical properties | Thermal expansion | | <ul style="list-style-type: none"> Atomic scale calculation of thermal expansion as a function of composition and defect content Mesoscale modelling of fuel thermal expansion |
| | Creep: thermal and under irradiation | | <ul style="list-style-type: none"> Atomistic modelling of high temperature and irradiation effects on the mechanical properties of fuels Atomistic modelling of fuel deformation behaviour High temperature creep experiments with controlled conditions oxygen partial pressure control in Measure creep under ion and neutron to evaluate the radiation induced creep component as a function of temperature and load |
| | FCMI | | <ul style="list-style-type: none"> Atomistic modelling of thermal expansion in presence of defects and fission products Measurement of thermal expansion of infused or self-irradiated fuels |
| Compatibility between fuel, cladding and coolant | Cladding corrosion: ROG | | <ul style="list-style-type: none"> Electronic structure calculations of thermodynamic data on compounds between FP (uranates, plutonates...) Synthesis of selected compounds between FP and fuel elements and measurement of thermodynamic data (melting temperature, thermal conductivity...) Development of thermodynamic database describing Gibbs energy functions of the phases of the (Cs-I-Te-Mo)-(U,Pu)-O system as function of temperature and composition Using this model, thermodynamic equilibria calculations to predict the phase formation in nuclear fuel with respect to oxygen potential and temperature Carbides, nitrides: study of carburization and nitridation reactions |

| | | | |
|--|---------------------|--|--|
| | Fuel/coolant issues | Fresh, self-ion or neutron irradiated MOX (U,Pu)O ₂ UO ₂ (U,Pu,Am)O ₂ | <ul style="list-style-type: none"> ▪ Electronic structure calculations of thermodynamic data on compounds between FP, fuel (U, Pu, Np, Am, Cm, C, N, O) and coolant (Na, Pb) elements ▪ Synthesis of selected compounds and measurement of thermodynamic data (melting temperature, thermal conductivity...) ▪ Development of thermodynamic database describing Gibbs energy functions of the phases of the (Cs-I-Te-Mo)-(U,Pu)-O system as function of temperature and composition |
| | Corium composition | (U,Am)O ₂ UC (U,Pu)C UN (U,Pu)N (Pu,Zr)N SIMFUELS (fresh fuels including FP) | <ul style="list-style-type: none"> ▪ Electronic structure calculations of thermodynamic data on compounds in the (U-Pu-Fe-O), (U-Pu-Fe-C) and (U-Pu-Fe-N) systems ▪ Synthesis of selected compounds in these systems and measurement of thermodynamic data (melting temperature, thermal conductivity, miscibility gap...) ▪ Development of a thermodynamic database for the (U-Pu-Fe-O) (U-Pu-Fe-C) and (U-Pu-Fe-N) systems |

3.2.3 Development of advanced fuels

The needs for longer term R&D concerning fuel and relative fabrication processes are identified in the following areas: (i) oxide fuels with optimized microstructure, including fuels incorporating Pu coming from multirecycling and transmutation fuels, (ii) advanced carbide and nitride fuels and (iii) alternative fuel concepts (see Table 10).

For each case dedicated irradiation testing will also be necessary in material testing or fast neutron reactors. Once removed from the reactor, PIEs must be performed.

• **Advanced oxide fuels development**

Powder metallurgy will still be the choice for first MOX fabrication plants, but co-conversion routes to generate MOX powder should be developed to reduce dust issues, enable optimised remote handling with degraded Pu vectors (and concomitant ²⁴¹Am pollution due to ²⁴¹Pu decay) when Pu multi recycling is established, and to achieve greater Pu homogeneity, enabling greater reliability and accuracy in properties, lower risk of hot spots under power, and a product more readily dissolvable to close the nuclear fuel cycle.

Incremental process improvements can be made. In particular, a deviation (partial or complete) from traditional powder blending should be considered to yield a more homogeneous MOX fuel devoid of Pu rich regions, enabling a more uniform burnup and a material readily soluble under PUREX conditions. Such an evolutionary step can be achieved at the conversion stage at the recycling plant, where the U and Pu solutions are unified in a predetermined Pu enrichment. The (U,Pu) solution can be co-converted into a powder ready for further manipulation and processing into product pellets. Classical hydroxide precipitation is to be avoided as it results in a dangerous material – ammonium nitrate, which is explosive when dry. Other co-precipitation routes should be considered, for example (U,Pu) oxalate precipitation. Plant simplification and automation will play an important role too. The implementation of such processes can be accelerated if important issues pertaining to dust can be eliminated. Ideally the co-precipitation steps should result in an essentially dust-free powder, and should in an optimum situation not contain particles with diameters less than 20 µm aerodynamic diameter. Such particles should also be spherical or near spherical to ease their transport in the production lines. If dust can be eliminated, the potential for radiological pollution of the gloveboxes and the equipment therein can be reduced.

The use of dopants to create advanced microstructures can also be envisaged (e.g. enhanced grain size and creep rates), as well as alternative fabrication routes for MOX pellet productions in the long term including, but not limited to, spark plasma sintering or additive manufacturing.

In the longer term, as described in section 2.2, Pu multi recycling in fast reactors must be developed. The increased presence of americium from increasing levels of ²⁴¹Pu will increase the radiation hazard necessitating improved plant designs and plant automation. Industry today has vast experience in automated

pellet fabrication in France and UK Melox and SBR plants. Despite their high level of automation, both are based on operator intervention (for maintenance and adjustment of equipment) using glovebox technology. The viability of this type of intervention needs to be evaluated for the future.

An important challenge to be overcome for the production of transmutation fuel will be the need for increased biological protection because of the increased Am and Cm content present in these fuels. It is likely that highly shielded hot cells will be needed for the production of such fuels. Intervention then would only be possible through engineered solutions to move equipment from the production lines into specially constructed facilities for repair or adjustment. These challenges will extend all the way to assembly production, heat removal during transport, and shielding at the reactor for storage before loading in the reactors.

- ***Carbide and nitride fuels development***

As indicated in section 2.2, in the long term, optimising core performance in terms of breeding and increased margins to melt would necessitate the adoption of mixed uranium and plutonium carbides and nitrides (MX = MC and MN). Their fabrication is not trivial, if high purities are to be achieved, and there are question marks about their volatility at temperatures below the melting point.

Advanced driver fuels of the future (MC and MN) have never been manufactured on the large scale, and there are many needs to be fulfilled. The tried and trusted carbothermal reduction (CTR) route used widely in the past to convert the oxide feed stocks to carbide or nitride by their thermal treatment when in intimate contact with carbon needs to be improved or replaced. This process is relatively simple, but the high temperatures result in powders with very low specific surface area, rendering them difficult to sinter. Communion steps are necessary, and with them the generation of fine highly pyrophoric powders. Improved fabrication methods must be envisaged to ease the production and improve the product quality. Electric field assisted sintering procedures such as spark plasma sintering (SPS) are being developed and offer the (as yet unproven) potential to yield high quality pellets without deleterious milling steps.

In the case of nitrides, enrichment in ¹⁵N during fabrication and its recovery from CO in the off gases needs to be dealt with. Due caution and respect must be given to controlled atmospheres in the gloveboxes, with 10 ppm in oxygen and water or less being essential at all times. Furthermore, contingency plans (e.g. nitrogen or argon flooding) must be elaborated to combat inadvertent rupture of the glovebox containment.

Beyond these evolutionary synthesis developments, revolutionary synthesis routes for carbides and nitrides are needed to shake off the undesirable attributes of the CTR procedure completely. These can be as simple as new routes to co-precipitate the carbon oxide precursor to reduce the CTR temperatures, or radical to generate the carbide and nitride powders via an organometallic route.

In addition, dedicated samples must be produced and the intrinsic properties of these materials determined to the highest accuracy, after which, the in pile behaviour of these fuel forms must be investigated, leveraging past knowledge with further dedicated experimental and theoretical programmes.

- ***Alternative fuels concepts (Molten Salt Reactor fuels/containment and materials)***

In addition to the conventional pellet-in-pin fuel designs of the ESNII reactors, also the fuel of the Molten Salt Reactor (MSR) designs will be investigated. MSR's generally use molten fluorides as fuel carrier of the fissile (U, Pu) or fertile (Th) elements, but also chloride salts are under consideration. The scientific and technical issues relevant to the behaviour of this type of nuclear fuel under irradiation are very different from solid fuels. Radiation effects play little role in the fuel behavior, thermal transfer of the fission heat is strongly coupled to the fluid dynamics, and solubility and retention of the fissile material and fission products in the molten salt are of major importance for the safety characteristics of the reactor. Since the MSR reactor concept is generally coupled to on-line clean-up of the salt to allow long term continuous operation, the separation of the fission products from the salt is integrated in the fuel research. The key research topics are summarized in **Table 10**.

Table 10: Main issues concerning the development of advanced fuel materials

| Main issue | Breakdown in sub-issues | Materials concerned | Techniques/ Methods |
|---------------------------------------|---|--|--|
| Advanced oxide fuels development | Microstructure, process and design optimisation | MOX | <ul style="list-style-type: none"> ▪ Development of advanced microstructures, increased homogeneity (e.g. chemical methods, improved milling, use of dopants) ▪ Alternative fabrication routes for MOX pellet productions: e.g. SPS, additive manufacturing... ▪ Design, geometry improvements. ▪ Complete characterization (fresh fuel, irradiation, PIE) |
| | Transmutation fuel development (homeogeneous and heterogeneous) | MA bearing MOX, IMF (Cer-Cer, Cer-Met) | <ul style="list-style-type: none"> ▪ Development of advanced safe fabrication processes (heavily shielded or in hot cells) ▪ Development of recycling strategies (Pu and MA multirecycling, recycling of non-actinides, e.g. Mo). ▪ Complete characterization (fresh fuel, irradiation, PIE) |
| Carbide and nitride fuels development | Fabrication and design issues | UC, (U,Pu)C, UN, (U,Pu)N, (Pu,Zr)N | <ul style="list-style-type: none"> ▪ Develop safe fabrication and recycling routes for different geometries (e.g. microspheres and/or pellets), considering radiological protection and pyrophoricity ▪ Develop innovative methods (e.g. chemical methods for MX powder synthesis, Spark Plasma Sintering...) ▪ Solve recycling issues (e.g. gaseous release during dissolution, ¹⁴C, Tritium, fuel dissolution, options for ¹⁵N recycling and/or reduce cost of ¹⁵N...) |
| | Assessment of safety and fundamental properties | | <ul style="list-style-type: none"> ▪ Assessment of thermal stability (Decomposition of nitrides, vaporisation (during processing), thermal stress cracking. ▪ Compatibility tests between fuel, cladding and coolant ▪ Evaluation of pyrophoricity and oxidation behaviour (thermochemical modelling of the reaction with air and moisture, microstructural studies of the oxidation mechanisms) ▪ Complete characterisation using thermo-physical, chemical, mechanical and microstructural characterisation techniques |
| Alternative fuels concepts | Molten Salt Reactor fuels/containment and materials | Molten salt mixtures | <ul style="list-style-type: none"> ▪ Synthesis and purification of halide salts ▪ Melting temperature determination, phase diagrams ▪ Solubility and activity coefficients of the actinides and fission products ▪ Thermal properties of the liquid phase (heat capacity, thermal conductivity, viscosity, density, surface tension) ▪ Retention capacity for fission products (I, Cs, Te) ▪ Interaction with other core materials ▪ Tritium management |

4. Cross-cutting issues

4.1 Objectives

The correct qualification of materials with respect to their behaviour in their environment and working conditions, the development of models capable of anticipating materials degradation in operation, as well as the development of new materials with targeted improved properties, are widely recognized through all energy technologies, nuclear and non-nuclear, to have a huge potential to enhance the safety, efficiency and lifetime of energy production devices and to contribute to lower their maintenance costs. The EERA JPNM firmly believes that cross-fertilisation on materials between different technologies can only be beneficial and contribute to accelerating progress. Furthermore, pursuing joint research on materials topics of common interest that require the use of similar methodologies and facilities, can be an effective way to make better use of available human, infrastructural and financial resources.

First and most naturally, many common or contiguous research topics exist with other nuclear fission technologies, namely GenII/III reactors and (very) high temperature reactors for co-generation, especially exposure to irradiation damage and subsequent characterisation, similar materials and common needs for facilities. Then, radiation damage and its characterisation are also of interest to fusion energy technology. The EERA JPNM therefore deployed significant effort to dialogue with the platforms involved, NUGENIA for GenII/III systems, NC2I for VHTR and EUROfusion for fusion energy, to identify cross-cutting issues. The connection with the two former platforms has been formally established through the signature of the MoU with SNETP, while in the case of EUROfusion the H2020 project M4F is the first example of a project jointly run by fission and fusion materials communities.

Moreover, commonalities exist with energy technologies outside the nuclear field. While each energy system faces different materials' challenges, cross-cutting issues with nuclear materials can be identified, in particular concerning metallic materials and especially steels that operate under extreme conditions, in our case high temperature and aggressive operational environments. It is believed that the competences developed in the nuclear field on materials can be of use also for other communities.

As a general principle, collaboration on materials qualification and development across low carbon energy technologies on cross-cutting issues is a way to accelerate the energy system transformation towards the highest possible standards of safety, reliability, increased efficiency and cost reduction.

4.2 Commonalities with GenII/III nuclear reactor materials

4.2.1 Integrity of structural materials

The first common area between GenII/III and GenIV nuclear technologies concerns the integrity (performance and ageing) of structural materials. Four topics for collaboration have been identified:

- Development and qualification of welding procedures, including the analysis of residual stresses
Failure in metallic components often occurs in welds, so the integrity of welds is critical for the safe performance of nuclear components. Among the different factors that impact the integrity of welds, residual stresses play an important role. Residual stresses in turn depend among others on the basic material properties and on the welding process. This issue is of major interest for both Gen II/III and Gen IV communities and austenitic steels are of interest as reference materials.⁵²
- Testing and qualification procedures for miniaturised specimens for both mechanical characterization and crack growth under environmental conditions
Miniaturised specimen testing would be beneficial to characterise mechanically neutron irradiated materials to limit activity handling issues and to optimize the use of the limited amount of neutron irradiated material generally available.⁵³ The issue as to whether these specimens can be used to predict real component behaviour is addressed in both communities. Validation of specimen geometry and testing procedure, undertaken by inter-laboratory exercises to achieve the standardization of the procedures, will benefit from an enlarged number of laboratories involved. The determination of the fracture toughness and crack growth rate is also of common interest.
- Advanced characterisation and multi-scale modelling of microstructural evolution under irradiation
Radiation hardening of steels may compromise component integrity. Hardening appears already after fractions of dpa round 300°C and increases further with dose, generally saturating only after several dpa. In steels, hardening is the primary driver of embrittlement. In both RPV steels and F/M alloys, impurities

⁵² See section 3.1.1, page 42.

⁵³ See section 3.1.1, page 43.

and solute atoms form radiation induced clusters rich in Si, Mn, P, Ni (and Cu). Nature, formation and evolution of these features depend on many variables, chiefly composition (influencing thermodynamic but also diffusion mechanisms of chemical species, point defects and their clusters) and irradiation conditions (temperature, dose and dose rate). Developing models to describe formation and evolution of these nano-features is accordingly the goal of several Euratom projects of both Gen II/III and Gen IV communities and could be addressed jointly.⁵⁴

- Ion irradiation as a neutron irradiation surrogate to gain better understanding of microstructural evolution under irradiation and improve identification of radiation resistant materials.

Due to the high cost of neutron irradiations and to the scarcity of facilities, irradiation experiments using alternative irradiation sources such as ions and charged particles in general, to contribute to materials screening and improvement, as well as to model development/validation, are clearly of interest for both communities. The issues to be addressed in terms of transferability of results between charged particle and neutron irradiation environments, i.e. specific PIE, development of models, etc., are common.⁵⁵ Collaboration will thus benefit both communities.

4.2.2 Fuel and cladding materials

A second area where close collaboration will be especially profitable for both GenII/II and IV communities is the research on nuclear fuels and claddings.

As far as qualification and assessment of procedures for safety and integrity are concerned, the following topics are of special common interest.

- Safety of oxide fuels

In the short term, research on oxide fuels and in particular MOX is of significant common interest. Needs are identified in: (i) measurement of safety relevant properties as a function of composition and burn-up⁵⁶, (ii) assessment of behaviour during irradiation, including post irradiation examinations (PIE)⁵⁶, (iii) continued improvement in safety in conventional synthesis technology⁵⁷.

- Innovative fuels and synthesis routes

In the longer term, increased sustainability can be reached through increased nuclear fuel recycling and MA transmutation, which require the development of innovative fuels and fuels allowing the burning of Pu and MA.

- Fuel performance codes development and validation

To meet future requirements it is essential that the fuel performance and safety codes are continuously improved and validated by reducing uncertainties and extending experimental data.

In addition, the further development of advanced mechanistic and multiscale modelling tools and the execution of separate effect experiments and detailed materials characterization⁵⁸, used in complement to technological research, is an area of common interest for both existing and innovative fuel designs fuel types.

Then, the materials and coatings that are considered as accident tolerant claddings for Gen II/III reactors because of high temperature and corrosion resistance are in some cases common to Gen IV future reactors e.g. silicon carbide composites (SiC_i/SiC), innovative steels (FeCrAl including ODS grades), etc. Main issues are standardisation, joining, and qualification in environment. Advanced fuels types (e.g. nitrides) and advanced fuel synthesis technologies for accident tolerant fuels are also areas to be addressed together.

Finally, the experimental facilities enabling the manufacturing and characterisation of fuels need to be maintained and expanded to support both the current fleet and the development of future reactors. It is therefore essential that support is provided jointly to irradiation facilities, hot cells and laboratories.

4.2.3 Innovative LWR Designs and Technologies

Three potentially common research areas are found also with the field of design of innovative LWR:

First, even though the Supercritical Water Cooled Reactor (SCWR) is not part of the ESNII portfolio and materials for the SCWR are not included per se in the EERA-JPNM activities, the SCWR is a Gen IV concept and material testing in contact with this type of coolant is consistent with the general need to study the corrosion behaviour of advanced materials, including welds and joints, developed for other environments, as well as to develop materials that are resistant to corrosion in hostile environment. In particular, it is likely that similar advanced material mitigation strategies will have to be envisaged, i.e. ceramic coatings, alumina-forming

⁵⁴ See section 3.1.2, page 47.

⁵⁵ See section 3.1.2, page 52.

⁵⁶ See section 3.2.1, page 67.

⁵⁷ See section 3.2.3, page 72.

⁵⁸ See section 3.2.2, page 70.

steels, etc.⁵⁹ Therefore, it is possible to foresee future joint projects in which the qualification of materials, in particular new material solutions, is extended to all coolants of interest for GenII/III and Gen IV, from liquid metal to gas, water and SCW.

Then, the development and application of advanced or novel materials manufacturing processes constitute important subjects for collaboration. Powder metallurgy offers advantages for the production of components for Gen II/III reactors and is also the technique used for the production of ODS alloys: there are thus opportunities for cross-fertilisation. Moreover, 9Cr ODS steels have improved resistance in SCW compared to conventional 9Cr F/M steels. There may also be the opportunity for involvement of industry in ODS alloy production. Another advanced/novel manufacturing method that could be of interest for NUGENIA and EERA-JPNM is additive manufacturing (3D printing). The latter allows manufacturing of components of complex geometry and is in particular interesting for small quantities. Projects devoted to similar issues cross-cutting through NUGENIA and EERA-JPNM could thus be considered in the future.

Finally, the development of new materials is a cross-cutting issue throughout the concepts. Materials more resistant to corrosion, in particular austenitic steels and Ni-base alloys, may be of interest for application not only in the SCWR, but also in water-cooled SMRs, given that in several of these concepts the fuel is envisaged to remain in the reactor until the end of its exploitation, without intervention for fuel replacement or reshuffling. This implies that sufficient guarantees need to be provided in terms of capability of the core materials, that keep together the fuel assemblies, not to fail due to, for example, irradiation-assisted stress corrosion cracking, for the whole reactor lifetime. In this framework, coatings, surface treatment (e.g. shoot peening) or alumina forming steels may be an interesting option to be explored.

4.3 Commonalities with materials for (V)HTR

The existence of subjects of common interest and common challenges related to materials between NC2I and EERA JPNM stems in particular from the strong similarities between GFR (ESNII system) and (V)HTR, which is the core business of NC2I: except for the fact that the latter includes the use of graphite as moderator, many components are similar or subjected to the same requirements, namely high temperature operation in contact with flowing pressurized helium. Therefore the materials of interest are largely the same.

High temperature resistant materials are being developed within the EERA JPNM for use either in the second phase of prototypes/demonstrators or in commercial GenIV reactors.⁶⁰ For intermediate heat exchangers, insulating structures, control rods, and other internal structures such as core restraints, core belts, core barrel and fuel cladding, these materials include ODS alloys and composite ceramic materials like SiC_f/SiC, as well as, to a lesser extent, refractory materials and Ni-based alloys. The latter (Inconel 617, Haynes 230, including here also Alloy 800 as high-Ni stainless steel) are the reference candidate materials for out-of-core high temperature gas-cooled primary system components of both GFR and V/HTR (intermediate heat exchanger), and some advanced grades such as Inconel 718 and 738, or advanced ODS alloys, are considered to match the requirements of the high temperature helium environment, as well as the severe mechanical load of gas turbine blades. For fuels, materials of common interest would be actinide carbides and nitrides.

In addition, all Generation IV reactors should be designed for a design life of 60 years. Thus the development of design rules and design curve and codification into design codes is a shared problem. The ASME BVPC code Section III Division 5 is now being revised to support the design of the US (V)HTR project and dedicated test programmes and development of associated design rules for 60 years design life are being implemented for a number of materials such as alloy 800H and 316 (austenitic stainless steels), Grade 91 (F/M steel) and Alloys 617 and 709 (Ni-based alloys).

The problems to be addressed jointly concern thus the degradation of the properties of the above materials in operation, due to the synergistic effect of high temperatures, mechanical stresses, radiation and gas coolant environments, accumulated over a long exposure time. Among them:

- Development and qualification of welding and joining procedures of high performance structural materials, including the analysis of residual stresses;
- Development of fabrication technologies for critical components;
- Investigation of mechanisms related to long term operation due to high neutron dose, such as swelling and irradiation creep, and/or under high temperature: thermal creep, synergies with dynamic strain ageing, etc.;
- Constitutive modelling and description of materials behaviour, from physics-based to engineering models, and damage development at high temperature, as basis for codification improvements and qualification;
- Micro/nano-sample testing for model validation and condition monitoring;

⁵⁹ See section 3.1.3, page 53.

⁶⁰ Issues mainly addressed in section 3.1.3.

- High-temperature stability for metals and refractory ceramics (thermal aging effects);
- High temperature materials' property database (tensile strength, creep, creep-fatigue, ...);
- Development and nuclear grade codification of composite and ceramic materials and the high temperature alloys.

Table 11 summarizes materials issues of potential cross cutting interest between EERA JPNM and both GenII/III (NUGENIA) and and VHTR (NC2I) communities.

4.4 Commonalities with fusion materials

Despite the significant differences that exist in terms of functioning principles and therefore design between GenIV fission and fusion energy reactors, a number of materials issues of common interest have been identified.

4.4.1 Classes of materials

Commonalities on materials concern almost exclusively steels. Fusion imposes stricter conditions than fission, due to the reduced activation requirement to minimise the impact of waste. Specifically, the high Ni content of austenitic steels, materials of choice for the design and construction of GenIV prototypes, raises insurmountable concerns in terms of activation for fusion applications. Austenitic steels are also much more prone to swelling and He production than F/M steels, especially under fusion conditions. Thus **F/M steels are the only possible choice as fusion reactor breeder blanket structural materials.**⁶¹ They can withstand the high neutron flux and energy of fusion (14 MeV), that leads to He production and swelling, up to >100 dpa, and can be produced in reduced activation grades (e.g. EUROFER in the EU, F82H in Japan, CLAM in China...). In the long-term F/M steels have a high potential as structural and cladding materials also for GenIV, in order to increase burnup. Therefore, **F/M steels are the only possible cross-cutting metallic material between fission and fusion.**

In terms of refractory materials, ceramic composites such as SiC/SiC have been long considered as structural materials for fusion, due to their especially low activation and excellent resistance to high radiation dose and temperature. They are still studied in some countries (e.g. Japan) as a necessary choice for advanced breeding blanket designs, beyond DEMO, so they could potentially be also a topic of joint research. At the moment, however, this material is not in the focus of fusion materials research in Europe.

Table 11: Summary of potential cross-cutting issues on materials through nuclear fission technologies

| General topic | Commonalities |
|---|--|
| Integrity of structural materials | <ul style="list-style-type: none"> • Development and qualification of welding procedures, including the analysis of residual stresses • Testing and qualification procedures for miniaturised specimens for both mechanical characterization and crack growth under environmental conditions • Advanced characterisation and multi-scale modelling of microstructural evolution under irradiation • Ion irradiation as a neutron irradiation surrogate to gain better understanding of microstructural evolution under irradiation and improve identification of radiation resistant materials |
| Fuel and cladding materials | <ul style="list-style-type: none"> • Safety of oxide fuels • Innovative fuels and synthesis routes • Fuel performance codes development and validation • Development of advanced mechanistic and multiscale modelling tools and the execution of separate effect experiments and detailed materials characterization • Materials and coatings for accident tolerant claddings • Experimental facilities and relevant problems of availability |
| Innovative Light Water Reactor Designs and Technologies | <ul style="list-style-type: none"> • Materials issues for SCWR • Development and application of advanced or novel materials manufacturing processes • Development of new radiation-, corrosion- and high temperature resistant materials |

⁶¹ D. Stork and al.: "Materials R&D for a timely DEMO: Key findings and recommendations of the EU Roadmap Materials Assessment Group", Fusion Engineering and Design 89 (2014) 1586–1594.

| | |
|------------------|--|
| (V)HTR materials | <ul style="list-style-type: none"> Issues related with development and qualification of high temperature resistant materials in gas coolant environment |
|------------------|--|

4.4.2 Compatibility with heavy liquid metals

The use of reduced-activation F/M steels in the blanket of a fusion reactor raises **corrosion** concerns because liquid PbLi eutectic is used in many blanket designs as flowing tritium breeder and in some designs also as coolant. To mitigate the effect of corrosion, temperature and liquid velocity need to be limited, but even in this case it is considered that **alumina or other protective SL are likely to be needed**. Even without contact with HLM, in fusion the surface treatment of steel structures is important to reduce tritium permeation through large thin-walled steel surfaces that separate coolant and breeding loops, which has to be avoided for tritium extraction efficiency and safety considerations. Rupture of the thin protecting layer that forms on the surface leads to increased tritium permeation, so its functioning and integrity is crucial. However, **the need for protective surface layers raises several questions** regarding the stability of the protection at the temperature of operation, especially under neutron irradiation, the possibility or not of self-healing (very limited or non-existent in the case of contact with PbLi), and the consequences of local rupture of the coating during thermal cycling. These problems are extremely similar to those faced by HLM-cooled systems in GenIV.⁶² Thus **protective surface layers for steels and their characterisation are clearly an issue of interest both for fission and fusion applications**. Moreover, the GenIV fission community has identified **LME of F/M steels** as a major concern, which is **expected to be a serious concern also for fusion**, although to date the fusion community has not given significant attention to this problem. Addressing the problem of LME, with a view to better understanding it and mitigating it, is thus a topic of high value for both the fusion and the fission communities, the latter being in this respect more advanced than the former.

4.4.3 Codification of F/M steels

The use of **F/M steels** for the design of both fusion and GenIV fission systems is hampered by the fact that these steels **are not yet fully included in the design codes** (e.g. RCC-MRx). Problems that are either specific to F/M steels (e.g. softening under cycling loads, low temperature embrittlement...), or common to all steels but especially serious for F/M steels (e.g. plastic flow localisation and subsequent loss of elongation after irradiation at low temperature) need to be addressed, based also on the understanding of the relevant physical processes. Different approaches from the case of austenitic steels, which are already codified, are needed. The codification of EUROFER is currently ongoing in the fusion community in view of the licensing of the test blanket for ITER by F4E. The fission community had also been involved in the codification of F/M steels, until these became materials of lesser priority. Many results obtained in the FP7/MATTER project turned out to be very useful for the fusion materials community. Hence **design rules for F/M steels, specifically cyclic softening and plastic flow localisation, can be considered as a common ground for the fission and the fusion materials communities**.⁶³ Linked with the process leading to the codification of F/M steels in general (and specific types of steel in particular) is also the usefulness of a **joint fission-fusion materials database**.

4.4.4 Welding procedures and characterization

In fusion components as well as in any other system component, welds are weak spot that require attention and are an integrating part of pre-normative materials research programme. Even though currently most attention on welds in GenIV is focused on austenitic steels and GenIV relevant environment, while in fusion F/M are the key class of materials, the qualification methodology can be common. **Certainly common are the issues of detecting defects in welds and evaluating their consequences, taking into proper account the presence of residual stresses**.⁶⁴ Moreover, any result concerning weldments on F/M steels and their response to irradiation and/or exposure to HLM will be equally valuable for the fusion and GenIV fission communities.

4.4.5 Small specimen testing

Establishing an accepted standardized methodology to extract component-relevant mechanical properties from sub-sized and miniaturized specimens is absolutely crucial for fusion. As a matter of fact, fusion reactions produce 14 MeV neutrons, thereby producing a spectrum that is significantly different from the fission spectrum and totally irreproducible in fission irradiation facilities. The impossibility of testing materials under fusion-relevant spectra constitutes a serious drawback for the establishment of fusion relevant design rules and for the licensability of fusion breeding blanket components. **Suitable neutron sources reproducing fusion-relevant spectra need to be built**, but these are costly facilities and so far no source of this type is operative.

⁶² See section 3.1.1, page 40, and section 3.1.3, page 64.

⁶³ See section 3.1.1, pages 38-42; section 3.1.2, page 48; and section 3.1.3, page 56.

⁶⁴ See section 3.1.1, page 42.

Even when such a source becomes operative, **the volume of irradiated material will be extremely small** (how small depends on the design and the ambition, and therefore the cost, of the source). Therefore **the use of sub-sized and miniaturized specimens becomes mandatory and their standardization represents another ground for collaboration between fission and fusion materials communities.**⁶⁵

4.4.6 Development of F/M steels with better high and low temperature properties

In the He-cooled DEMO design, as well as in commercial GenIV fission reactors, temperatures in excess of the current limit of 550°C for F/M steels are targeted to increase thermal efficiency and so optimise the use of resources. For this purpose, the creep resistance of F/M steels becomes the property of concern that needs to be improved. Moreover, the low temperature embrittlement under irradiation is an unavoidable concern since the blanket of the water-cooled DEMO design is planned to operate between 250 and 350°C. This will make the problem of low temperature embrittlement of F/M steels into a key issue to be resolved, even more than for GenIV systems. Thus, **both advanced GenIV systems and fusion share the important goal of improving the creep resistance of F/M steels to above 650°C and of reducing their susceptibility to low temperature (<350°C) embrittlement.** This implies the development and qualification of industrially-scalable innovative materials such as ODS F/M steels or conventional F/M steels with enhanced creep strength. The thermomechanical treatment and composition tuning are also expected to improve the ductility, with an impact on reducing low temperature embrittlement.⁶⁶ While fusion is subject to the stricter requirement of reduced-activation, the cross-fertilisation for what concerns the criteria used for the tuning of the composition and certainly for the choice of TMT is certainly going to be fruitful.

4.4.7 Tools to screen among radiation-resistant materials

Irrespective of the issue of the very specific fusion neutron spectrum, most of the characterization of fusion materials can only be done using fission facilities. However, also for fusion neutron irradiation experiments are costly and the irradiation facilities are scarce. Moreover, for fusion even more than for fission there is a need to produce optimized materials. To date the materials options for DEMO remain very open, given that even Eurofer, which is the current reference reduced activation F/M steel, is likely not to be the final material that will be used for the construction of DEMO, if F/M steels with better high and low temperature properties are developed (see above). Therefore, tools to screen among different materials in terms of, in particular, response to irradiation, are as much needed for fusion and fission applications. As in generation, these rely on the **combination of charged particle irradiation, coupled with suitable PIE and, in particular, mechanical property probing techniques on small volumes** (see above). The development of an **established methodology** to achieve the goal of characterizing materials using these tools is thus yet **another ground for collaboration between fission and fusion materials research communities.**⁶⁷

4.4.8 Advanced modelling

The elaboration of robust rules defining materials behaviour, the identification of appropriate mitigation strategies and the improvement of materials properties depend crucially on the precise understanding of the physical processes that govern the evolution of materials exposed to extreme conditions. Thus, **the continuous improvement of the level of understanding of materials behaviour by exploiting advanced microstructural characterization and evolving physical models is clearly a common issue and goal both for fission and fusion applications.** Modelling activities of specific common interest may concern: microstructure evolution under irradiation, low temperature hardening and embrittlement, irradiation creep and swelling, liquid metal corrosion and liquid metal embrittlement, ODS fabrication processes, precipitation coarsening and recrystallization of oxides or carbides in advanced steels.⁶⁸

Table 12 summarizes materials issues of potentially cross-cutting interest between GenIV fission and fusion.

⁶⁵ See section 3.1.1, page 43.

⁶⁶ See section 3.1.3, page 55.

⁶⁷ See section 3.1.2, page 51.

⁶⁸ See section 3.1.2.

Table 12: Summary of GenIV fission/fusion potential cross-cutting issues on materials

| General topic | Commonalities |
|---|--|
| F/M steels for current concept designs | F/M steels <ul style="list-style-type: none"> • Design rules in RCC-MRx for F/M steels • F/M HT assessment: plastic flow localisation, cyclic softening, thermal creep, thermal fatigue, creep-fatigue, • F/M welding • F/M compatibility with LM: corrosion, erosion, LME |
| Innovative high temperature resistant steels | <ul style="list-style-type: none"> • Optimisation of F/M ODS & TMT steel fabrication <ul style="list-style-type: none"> - Identification of best compositions and TMT - Deformation modes of ODS alloys at high temperature (creep) - Stability of the microstructure after long exposure to high temperature and long time for the ODS &TMT steels • Screening methods for prospective new materials (including charged particle irradiation and small specimen testing technology) |
| Ceramic materials | <ul style="list-style-type: none"> • Ceramic coatings on steels (alumina or aluminium containing) |
| Physical modelling and modelling-oriented experiments | F/M steels <ul style="list-style-type: none"> • Microstructure evolution under irradiation • Low temperature hardening and embrittlement and plastic flow localisation • Irradiation creep and swelling • Liquid metal corrosion and liquid metal embrittlement • ODS fabrication processes, precipitation coarsening and recrystallization of oxides or carbides in advanced steels, ... |

4.5 Commonalities with materials for other energy technologies⁶⁹

4.5.1 General methodological common patterns

An analysis of materials needs and materials science approaches through energy technologies in EERA, including nuclear, led to the identification of the following common patterns⁷⁰:

- **Ageing and degradation mechanisms studied by combining advanced experimental characterization with multiscale modelling.** This approach is ubiquitously used and perceived as the only way to really advance towards extending the lifetime of materials and/or improving their performance. Even if the problems addressed are different depending on the energy technology, in most cases the (experimental and modelling) techniques and the approaches used will be similar, being it also likely that similar methodological problems need to be addressed and solved, irrespective of the specific type of materials or issues studied. In particular, a very general problem that can be jointly addressed concerns the establishment of protocols for the application, analysis and comparison with simulation results of microstructural examination techniques⁷¹. The problem of accelerated testing (key to address long-term ageing) partly enters this topic, too, because it may be solved by combining specific testing techniques with models that allow the safe extrapolation to much longer exposure times.
- **Characterization of energy materials and devices: contribution of large scale facilities, as well as *in situ* and *operando* techniques.** The use of large scale facilities for materials exposure and testing/characterization is a need through all energy technology materials. In particular, *in situ* and *operando* characterization techniques have been identified as key for studying materials in working

⁶⁹ This part is largely based on the outcome of two workshops: (1) the EERA inter-JP cross-fertilization workshop on materials for energy applications and technologies, held on April 28th-29th, 2015, in Brussels [http://www.eera-jpnm.eu/files/sharer/documents/JPNM-related_Meetings/2015_04_28-29_EERA_InterJP_Workshop_Energy_Materials]; (2) the workshop on cross-cutting issues in structural materials R&D for future energy systems, held on November 25th-26th, 2015, in Petten, NL [http://www.eera-jpnm.eu/files/sharer/documents/JPNM-related_Meetings/2015_11_24-25_MatISSEXcuttingWshop_Petten]. Moreover, an initiative on high temperature materials through energy technologies is ongoing, that involved several EERA JPs and should lead to a joint EERA position paper on the subject.

⁷⁰ http://www.eera-set.eu/wp-content/uploads/EERA-JP-workshop-Materials_for_Energy_report.pdf

⁷¹ See section 3.1.2.

devices, although the requirements and possibilities to actually do so heavily depend on the energy technology. In the case of nuclear materials, *operando* techniques have currently limited application but it may be very interesting to interact with other energy technologies in connection with the issue of materials controllability.

- **Rational design of materials supported by modelling.** For all technologies there is a need to find new or alternative materials to improve the performances and the durability of the materials and devices, to decrease their costs or to avoid critical or hazardous elements. Examples are non-noble metal based catalysts, or photovoltaic cells free of hazardous elements, such as cadmium or lead. This calls for an approach based on rational design supported by modelling, which is common to all energy technologies. The extent of applicability of this approach will be different in each case and will depend on the complexity of the phenomena involved and how the adequate functionality of specific materials can be quantified.

4.5.2 Materials for high temperature applications

In addition to the above common general patterns, resistance to high temperature has been recognised as a requirement for materials in a wide spectrum of energy technologies. First, irrespective of the origin of the heat, the efficiency of thermodynamic cycles operating between two heat reservoirs is improved by increasing the temperature of the hot one. Moreover, some energy production systems simply inherently require high temperature in order to function because of the physical-chemical processes involved. Thus, **low carbon energy technologies as different from each other as fuel cells and hydrogen, concentrated solar power, bioenergy, geothermal, GenIV nuclear fission, and fusion, find commonalities in the need to operate at temperatures above, and sometimes well above, 400°C.** During the transition to a fully low-carbon economy, this problem affects also clean fossil fuel plants, which will as well need to operate at the highest temperature possible to increase efficiency and minimize emissions. In this context it is worth mentioning that The European Creep Collaborative Committee (ECCC)⁷² was formed in 1991 to co-ordinate Europe-wide development of creep data for high temperature plants. ECCC has probably the largest and most complete set of creep data in Europe for parent materials and welds and issues regularly guidelines on data generation and assessment methods. In 2017 France decided to rejoin after 10 years absence and ECCC started a new activity for nuclear applications focusing on 316SS and long-term creep.

Temperatures in excess of 400°C, approaching and sometimes even exceeding 1000°C, together with thermal cycles imposed by power fluctuations, inflict severe thermomechanical stresses on the plant components. This requires the use and development of materials that should be proven to maintain their integrity and properties for a sufficiently long time at high temperature and/or when subject to thermomechanical fluctuating loads, within reasonable costs. Moreover, high temperature operation requires efficient cooling, leading to the production of superheated steam or to the use of alternative coolants like liquid metals, molten salts or gases, to which materials will have to be exposed. Environmental degradation due to salt water also affects other technologies, such as off-shore wind turbines or ocean energy. Thus materials need to be also resistant to the attack of these environments.

The components that are mainly affected by high temperature and environmental degradation problems are, quite clearly, those where heat exchange and transfer takes place, either physically or chemically, like boilers, turbines and pipes, in addition to absorbers, receivers, electrodes and catalysts... Some of these components have structural functions, their integrity having therefore also safety implications. Others need to maintain their properties mainly for system efficiency, and therefore also economic, reasons.

The materials selected and qualified to demonstrate that they are able to maintain their properties at high temperature and in aggressive environments, also need to be joinable or weldable, fabricable, formable and - importantly for many renewable forms of energy- low cost.

Almost irrespective of the energy technology considered, only a few classes of materials can potentially meet all these requirements, with different levels of quality in their response:

- Some materials are currently used or are anyway commercially available, but need to be demonstrated to be suitable for the application. These are: creep-resistant F/M steels; austenitic steels; Ni-base alloys; and ceramics, mainly silicon carbide composites (SiC/SiC) and alumina based ceramics. Often, they need appropriate coatings as protection from environmental aggression.
- Other are incremental improvements on the above materials that offer potential for better efficiency or functionality of the system/component. These are developed to different TRL and are: CSE F/M steels that may operate up to 670°C and ODS steels for operation above 700°C; protective and durable coatings suitable to withstand cyclic operation or alumina-forming austenitic (AFA) or ferritic (FeCrAl) steels;

⁷² <http://www.ommi.co.uk/etd/eccc/advancedcreep/com.html>

advanced composites (SiC_i/SiC with improved oxidation/corrosion stability, Al₂O₃/Al₂O₃, mullite/mullite...). Importantly, for these new materials joinability and industrial production should be ensured.

- Finally, prospective materials are: oxide ceramics with infiltrated nano-catalysts; refractory metallic alloys (V, Mo, W...); MAX phases.

This list of materials proves once again to have much in common with GenIV structural materials.

In general, structural materials qualification in environment and new materials development are costly activities. The quality requirements imposed on nuclear materials are very strict and the conditions they are expected to face, that combine irradiation, environmental degradation and thermal loads, are especially severe. These facts led over the decades to the establishment of advanced competences in materials qualification and development in the nuclear field that may be beneficial for other energy technologies, where cost constraints related to low energy density or intermittent energy production, and therefore low profitability, do not allow sufficient investments for the development of new or better materials. Thus joining forces with other energy technologies to set up joint projects on high temperature materials represents a win-win situation.

4.5.3 Specific commonalities with concentrated solar power

Specific commonalities can be identified and targeted between materials issues for GenIV fission energy and concentrated solar power (CSP). Current large CSP solar tower systems, which are at a commercial stage, are still affected by the limited operation of the ceramic solar receivers, which would largely benefit from the use of SiC_i/SiC to extend life and working temperatures above 800°C (air temperature).⁷³ Different CSP systems are being developed, from "large tower" systems to small residential ones: it is the limited reliability of ceramics materials for the receiver that hinders the commercial success of these systems, especially large "solar tower" systems (>5MW_e), which are otherwise at an advanced maturity level after over a decade of intense R&D and are poised to achieve 20 years of maintenance-free operations. Moreover new receivers, made of composite ceramics (or ODS steels) with increased resistance to oxidation and thermal shock, would boost innovative smaller "dish systems", which should withstand working temperature of about 900°C to be coupled directly to a gas micro-turbine. Nuclear materials can also contribute to the development of exchange thermal fluid suitable for heat storage above 600°C. A credible solution rests on the deployment of liquid lead, which would benefit from existing GenIV knowledge and technology for LFR.

Table 13 summarizes the cross-cutting issues through low carbon energy technologies (fuel cells and hydrogen, concentrated solar power, bioenergy, geothermal, GenIV nuclear fission) concerning high temperature, environmental degradation resistant materials.

Table 13: Summary of cross-cutting issues on high temperature and environmental degradation resistant materials through low carbon energy technologies

| General topic | Commonalities |
|--|--|
| High temperature (>400°C) mechanical performance assessment | <ul style="list-style-type: none"> • Resistance to • Thermal creep deformation (rupture beyond lifetime) • Thermal-mechanical fatigue (resistance to crack initiation/propagation) • Creep-fatigue interaction |
| Protection from aggressive environment (liquid metals, molten salts, gases, ...) | <ul style="list-style-type: none"> • Corrosion/oxidation/dissolution/erosion processes • Coatings of proven stability • Self-healing surface protection mechanisms |
| Other properties to be maintained | <ul style="list-style-type: none"> • Thermal conductivity and limited thermal expansion • Non-permeability (to specific elements, e.g. gaseous like H or He) |
| Steels for high temperature applications: existing and advanced | <ul style="list-style-type: none"> • Creep-resistant F/M steels (including ODS) • Austenitic steels • Ni-based alloys |
| Refractory materials: metals and ceramic | <ul style="list-style-type: none"> • V-, Mo-, W-based alloys • SiC_i/SiC • Alumina based ceramics • Max phases |

⁷³ SiC_i/SiC is also experiencing renewed interest in the aeronautical industry amid technological advances in jet turbines that could substantially reduce its costs.

| | |
|---|--|
| composites: existing and advanced | |
| Materials qualification | <ul style="list-style-type: none">• Exposure facilities• Test standardization |
| Advanced modelling and characterization | <ul style="list-style-type: none">• Physical mechanisms |

5. Timeline and milestones

The development and qualification of materials for any technology is a continuous process:

- The return of experience from the use of already qualified and licensed materials provides further input for the design codes and triggers the search for more suitable materials.
- Fundamental research on materials behaviour mechanisms provides tools to better assess component lifetimes and suggests routes to materials property improvement.
- Following the return of experience from use and the continuously improved understanding of the processes that govern materials behaviour, new routes to materials fabrication and processing and new materials design strategies are explored.

The virtuous circle of Figure 2 represents well this continuous process. Moreover, along the route leading to materials qualification for their use in a given technology unexpected results may lead to identify the need to introduce mitigation strategies that are not based on design solutions, but rather on materials science patches (this is typically the case when barriers or coatings are required), or even to discard certain classes of materials, obliging different possibilities to be pursued. These incidents and changes of routes can be very specific for a given technology and the relevant timeline and milestones may require very specific adaptations and frequent revisions.

It is therefore challenging to foresee in detail the timeline to the qualification and development of materials for advanced nuclear systems in a broad way, without reference to not only specific systems, but also specific components. Drawing such a detailed and system-specific roadmap in terms of timeline and milestones is, on the other hand, out of the scope of the **present document**, that **addresses the major materials research issues in global and largely cross-cutting way**. Yet, time constraints exist, that are related to the need to proceed with the design, licensing and construction of the systems. Thus, priorities and timeframes need to be given.

Accordingly,

Table 14 provides a **simplified timeline**, defined on the base of the need to have sufficiently qualified materials for the purpose of design and licensing of the different systems (milestones), allowing for the different TRL of both systems and materials. Each row refers to experimental work and supporting models aimed at a specific goal, as expressed. Clearly, the length of the arrows projects the goal to a farther date, thus *the shorter the arrow, the higher the priority*. At the same time, if the goal has to be reached on time, activities should start already now (in most cases have started already), but the farther the corresponding milestone, the more the relevant activities should be considered as *underlying technology research*, rather than high priorities. In some cases, if the goal is not reached as closer milestone, the activity remains of interest for farther milestones: this is indicated by an empty arrow with dashed outline. Goals that are somehow a pre-requisite to the main one are indicated as lighter colour arrows: the length is in this case shorter but arbitrary, and does not refer to any given milestone.

This chart is of course only indicative and will require progressive updating over time.

Table 14: Indicative timeline and milestones for the main activities outlined in the present strategic research agenda.

| | 2025 | 2030 | 2035 | | 2040 | 2050 |
|---|---------------------|--------|-------------------------|----------------|------|------------|
| | ASTRID I / MYRRHA I | ALFRED | ASTRID II / (MYRRHA II) | ALLEGRO I / II | FOAK | GenIV NPPs |
| Austenitic steels | | | | | | |
| Long term HT + low flux irradiation + He production effects (60 YDL) | | | | | | |
| LBE effects (incl. irradiation) on base material & welds | | | | | | |
| Lead effects (incl. irradiation) on base materials & welds; qualified corrosion protection (if | | | | | | |
| Advanced austenitic steels: swelling-resistant (AIM2), corrosion and creep resistant (AFA steels) | | | | | | |
| HT and flowing He qualified austenitic steels / AFA steels? | | | | | | |
| Ni-base alloys | | | | | | |
| HT and flowing He qualified alloys (secondary circuit. heat exchanger) | | | | | | |
| F/M steels | | | | | | |
| Long term HT + irradiation effects (current grades 91) | | | | | | |
| Qualified creep resistant F/M steels: ODS and/or CSE | | | | | | |
| Qualified creep and corrosion/LME resistant F/M steels: FeCrAl-ODS? | | | | | | |

| | 2025 | 2030 | 2035 | 2040 | 2050 | |
|--|---------------------|--|-------------------------|---------|------|------------|
| | ASTRID I / MYRRHA I | ALFRED | ASTRID II / (MYRRHA II) | ALLEGRO | FOAK | GenIV NPPs |
| SiC/SiC | | | | | | |
| Test standardization of continuum fiber ceramic composite cladding tubes | | This activity is largely a prerequisite for the next | | | | |
| Qualified SiC/SiC clad: hermeticity, thermal conductivity (before & after irradiation), corrosion resistant/protected, qualified joining | | | | | | |
| Refractory alloys | | | | | | |
| Radiation effect qualified and corrosion protected refractory alloys: Mo-ODS? | | | | | | |
| Perspective materials | | | | | | |
| Qualified HFA for specific application after screening among different possible compositions | | | | | | |
| Qualified MAX phases for specific application after screening among different phases / use as coatings | | | | | | |
| Fuel materials | | | | | | |
| Qualified FR MOX (U,Pu)O ₂ | | | | | | |
| Qualified MA bearing oxide fuels | | | | | | |
| Qualified advanced MX fuels (carbides, nitrides) | | | | | | |
| Qualified alternative fuel concepts (e.g. MSR fuels) | | | | | | |

6. Infrastructures for nuclear materials R&D

The research activities described in sections 3 and 4 have as essential prerequisite the availability of suitable facilities and infrastructures for materials qualification through exposure to conditions representative of the operational ones and subsequent characterization, both mechanical and microstructural. Modelling, on the other hand, implies the availability of suitable computational facilities.

The three service conditions of interest for GenIV reactor nuclear materials are **high temperature, contact with coolants and a high level of irradiation**. While the first two can be of interest and applications also for non-nuclear technologies (e.g. facilities for creep and corrosion tests, slow strain rate tests in environment...), **irradiation is quite specific of nuclear materials**. Therefore this section focuses mainly on facilities for exposure to irradiation and handling of irradiated materials.

6.1 Neutron Irradiation Facilities

6.1.1 Fast Neutron Facilities

Neutron irradiation facilities are a central necessity for performance and safety testing of all types of structural and fuel materials. Since the ESNII prototypes and demonstrators are all fast neutron spectrum systems, materials for their construction should be qualified in fast neutron irradiation facilities. Unfortunately, there is currently no fast neutron power or testing reactor operating in Europe. Worldwide, fast neutron testing reactors are limited to BOR-60⁷⁴ in Russia, CEFR⁷⁵ in China, FBTR⁷⁶ in India and JOYO⁷⁷ in Japan, the first one being the only one exhibiting significant availability. Russia also hosts two power SFRs, BN-600 and BN-800, running on MOX fuel, but these are in principle not available for materials qualification experiments, even if some fuel tests could be envisaged, for example Astrid fuel is expected to be tested partly in BN-600 or 800. To redress the situation, a fast system should be built in Europe, but this leads to a vicious circle because fast neutron data are needed for this purpose. This means that **at the moment Europe is totally dependent on non-European countries to have access to fast neutron flux facilities**. The construction of at least one of the ESNII prototypes within the coming two decades is therefore also essential in order to provide Europe with a facility in which structural and fuel materials for future commercial GenIV reactors, including innovative ones on which limited or no return of experience exists, can be fully qualified.

6.1.2 Material Testing Reactors

Materials testing reactor (MTR) are high power research reactors that can be used to test the behaviour of nuclear materials in operational conditions. Flexibility and ability to adapt to changing needs is a fundamental operating principle for such reactors. Material testing reactors can perform a variety of test irradiations simultaneously, the number of these depending on the MTR design and on the nature of the tests. In specific cases, MTRs can be augmented by loops emulating the coolant flow conditions in power reactors, although in practice devices of this type are very limited in existence and availability – and none exist in a fast spectrum. Almost all MTRs have on-site ancillary hot cells to make preliminary or detailed examinations, as well as to enable packaging and distribution of samples to other laboratories. Among the approximately 40 MTRs currently operational in the world, Europe hosts seven (see Table 15), the most prominent ones being the High Flux Reactor (HFR)⁷⁸ at Petten (The Netherlands); the BR2 reactor⁷⁹ in Mol (Belgium) and the HBWR⁸⁰ at Halden (Norway). These MTRs can be used for fuel testing and have appropriate hot cells available. Other reactors that currently do not have a license to handle MOX fuel, but are available for structural materials investigations, are LVR-15⁸¹ at Řež (Czech Republic), including the experimental loops for in-pile material testing under PWR/BWR, but also SCWR and high temperature He conditions, MARIA⁸² at Świerk-Otwock

⁷⁴ <http://www.niiar.ru/eng/node/224>

⁷⁵ <http://www.ciae.ac.cn/eng/cefr/index.htm>

⁷⁶ <http://www.igcar.gov.in/romg/fbtrdesc.htm>

⁷⁷ <https://www.jaea.go.jp/04/o-arai/joyo/english/joyo/roshin.html>

⁷⁸ http://iet.jrc.ec.europa.eu/sites/default/files/documents/brochures/hfr_mini_blue_book.pdf

⁷⁹ <https://www.sckcen.be/en/Research/Infrastructure/BR2>

⁸⁰ <https://www.ife.no/en/ife/laboratories/hbwr>

⁸¹ <http://cvrez.cz/en/infrastructure/research-reactor-lvr-15/>

⁸² <https://www.ncbj.gov.pl/en/o-nas/maria-research-reactor>

(Poland), and the TRIGA II reactors at Pitesti (Romania)⁸³ and Ljubljana (Slovenia)⁸⁴. In addition, materials still remain to be analysed from experiments performed at the Osiris reactor (Saclay, France), which was shut down end of 2015. The Jules Horowitz Reactor (JHR) is under construction at the CEA Cadarache site (France) to replace the capacity lost due to the shut-down of Osiris. **Europe's neutron irradiation capacity is currently very limited and even with the soon to start JHR, should no further new construction occur, the opportunities to test new materials in reactors will be restricted to the extreme.**

Table 15: Materials Testing Reactors currently operational in Europe and their irradiation characteristics.

| Name | Location | Maximum fast neutron flux (>0.1 MeV) [10^{14} n /cm ² s] (from ⁸⁵) | Dose rate [dpa _{Fe} /fpy] | Accessible temperatures [°C] |
|---------------|-------------------------|--|------------------------------------|------------------------------|
| BR2 | Mol, Belgium | 7 | <3 | 50-800 |
| HFR | Petten, The Netherlands | 5.1 | <7 | 80-1100 |
| HBWR | Halden, Norway | 0.8 | | |
| LVR-15 | Řež, Czech Republic | 3 | ~1 | |
| MARIA | Świerk-Otwock, Poland | 1 | | |
| Triga Pitesti | Pitesti, Romania | 1.8 | | |
| Triga Mark II | Ljubljana, Slovenia | 0.06 | | |

6.2 Charged particle irradiation facilities

For irreplaceable component materials, the doses expected at the end of life can be reached in existing MTRs, although in order to have exposure times much shorter than the real ones the fluxes used are much higher. While this may be a problem especially when the operating temperatures are high and the synergy between mild irradiation and thermal ageing and/or thermal creep may lead to unknown effects, MTRs available in Europe are at least usable to reach end-of-life doses, with the only caveat of the different neutron spectrum. However, for fuel and replaceable component materials, especially cladding, the doses expected in service are much higher than those accessible in thermal MTRs. The only way round this problem to reach doses up to or in excess of 100 dpa is currently the use of charged particles, despite the shortcomings and limitations associated with this type of irradiation, which can certainly not be used for full qualification purposes. Charged particle irradiation is also used to get further insight into the behaviour of materials under irradiation (see sections 2.3.2, 3.1.2 and 3.2.2)

A significant number of facilities permitting charged particle irradiation, which can be exploited for the above purposes, exist in Europe at nuclear research centres and universities.

Concerning fuel, if ion irradiation is used to study uranium-bearing fuel behaviour under irradiation, no charged particle irradiation facility exist for plutonium containing materials in Europe, and even in the world, except the Casimir facility at CEA Saclay (France)⁸⁶. This facility, however, is dedicated to the ion beam analysis of radioactive materials, and can only be used, with some limitations, to irradiate materials using light ions (H, He, N...). In practice, fuel pins are only qualified for a limited amount of time sufficient to license the first core. Further qualification or the qualification of alternative fuel pins are then performed in the reactor itself, when built.

6.3 Hot Cells and shielded facilities

Materials exposed to neutron irradiation need to be handled, tested and examined in dedicated hot cells or similarly shielded facilities. In particular, capsules taken out of the reactors need to be safely dismantled and hot cells on the sites of all European MTRs are an integrated component of any experiment.

⁸³ <https://www.nuclear.ro/en/departments/triga.php>

⁸⁴ <http://www.rcp.ijs.si/ric/description-a.html>

⁸⁵ <https://nucleus.iaea.org/RRDB/RR/ReactorSearch.aspx>

⁸⁶ http://iramis.cea.fr/Phocea/Vie_des_labos/Ast/ast_sstechnique.php?id_ast=361 (in French only)

For specimen fabrication and large specimen testing, e.g. mechanical testing, dedicated hot cells are absolutely necessary. For microstructural specimen fabrication and characterization, or in other specific cases where the quantity of material to handle is small, simpler types of shielding are sufficient. But invariably these are **very costly infrastructures**, both to build and maintain, and are available and employed only in a few equipped and licensed research centres, under severe safety rules for the operators, and only very rarely in universities.

The main hot cell facilities in Europe are located next to the European MTRs (see section 6.1.2), as well as at the CEA centres of Saclay, Cadarache and Marcoule (France), at the Sellafield site of the UK National Nuclear Laboratory, (NNL) and at Joint Research Center in Karlsruhe (European Commission). The largest part of this hot cell capacity was built in the 1970s and 1980s and despite regular refurbishments, might not be able to be operated much longer, because of the increasing maintenance costs and increasingly demanding safety regulations for this type of facilities. Two hot cell facilities dedicated to fuel located in Cadarache (LEFCA and LECA) will close in the next few years for these very reasons, while only one full scale replacement is planned to be built: the MOSAIC hot laboratory, which is planned to be built in Cadarache (France) to replace both the LECA fuel hot lab (Cadarache) and the LECL laboratory for structural materials (Saclay) facilities, when these close - the funding of this project, however, is not yet guaranteed. JRC is modernising part of the infrastructure with the construction of a new laboratory space (Wing M), to replace partially the aged facilities. Only the UK has known a recent significant increase of its hot cell capacity, with the complete refurbishment of the Windscale Laboratory and the construction and commissioning of the Phase 1 and of part of the Phase 2 of the Central Laboratory, both at Sellafield. In addition, a new hot cells complex has been recently built at CVR (Czech Republic): it contains 10 individual hot cells interconnected by the transport system. Two types of cells are utilized, gamma cells designed for testing the fuel cladding and those for irradiated nuclear power plant structural materials. They can accommodate up to ~300 TBq activity converted to ⁶⁰Co isotope. The alpha cells are designed for R&D on radioactive waste processing. Finally, the GENESIS platform in France enables characterisation of irradiated materials: at CEA/DMN Saclay (for highly active materials –in the CEA LECL laboratory, so long as it operates) and at CNRS/GPM Rouen, France (where new installations have been built, able to host activity up to 200MBq/sample). GENESIS is thus a new instrumental platform for nanoanalysis effects of radiation in materials (FIB-SEM, APT, HRTEM).

The limited availability of hot cell facilities, especially those licensed for fuel handling, whose availability has declined steadily over the decades, limits enormously the number of tests, measurements and examinations that can be performed on irradiated materials. This determines the fact that complete post-irradiation examination (PIE) of samples from irradiation experiments may take several years to be completed. This problem is exacerbated by the ever rising costs and difficulties of irradiated materials transport and the fact that current PIE include advanced microstructural examination with new techniques that did not exist when most hot cell facilities were built in Europe. In the case of fuel, the limited hot cell capacity also has important negative consequences on Europe's capability of fuel procurement, since these facilities are indispensable for the preparation and safety testing of new fuel.

Despite fair availability, Europe's aged hot cell capacity is already insufficient today and, as they age further, their replacement will be imperative. Given the time and budget necessary to build and commission a hot cell facility, new replacements should be planned right now at national and/or European level.

6.4 Transportation of samples for dedicated analysis

Suitable, flexible transport flasks and containers for irradiated test specimens are essential to make an effective use of Europe's current geographically separated test reactors, irradiation facilities and hot cells, as well as of the specificities of the various facilities. The capability to ship irradiated materials to and between appropriate facilities is an essential component of European infrastructure. Currently, in addition to the inherently high costs of this type of shipments, the existence of differences in the laws and rules applied to radioactive material transport through European countries often causes this type of transports to suffer from unduly very large delays, which effectively increase the overall costs, reduce the effectiveness of the research and prevent the timely completion of projects. **Improved coordination and standardisation of regulations and transport containers are here essential to reduce transport times and costs, and to break the strangle hold of ever increasing costs for fewer transports.** A parallel effort can be done in miniaturising the size of the samples to be transported for specific analysis, for example by employing FIB.

6.5 Computational facilities

Even if the need for computing facilities is not specific to the investigation of nuclear materials, it is worth stressing that the ICME approach described in section 2.3.2, as well as the simulations using design and fuel performance codes, are computationally intensive and call for access to world class high performance computing (HPC) systems.

More generally, this access is essential for international competitiveness in science and engineering. The importance of developing HPC capacity has been recognized by countries such as USA, Japan or France since the 1990's. Europe has recently acknowledged that HPC was a strategic resource for Europe's future and the need for a European-level policy to optimise national and European investments and adopted its HPC Strategy in 2012. It combines 3 elements:

- Developing the next generation of HPC technologies, applications and systems towards exascale HPC,
- Providing access to the best supercomputing facilities and services for both industry (including SMEs) and academia complemented with training, education and skills development in HPC,
- Achieving excellence in HPC applications in domains that are most important for Europe.

Concerning the access to supercomputing facilities, the vision is that of a pyramidal ecosystem, with at the top 3 to 5 European-level supercomputing centres (Tier-0), supported by a Tier-1 composed of national centres, in turn supported by a Tier-2 of regional or "meso"-centres.

Industry and SMEs are increasingly relying on the power of supercomputers to come up with innovative solutions, reduce cost and decrease time to market for products and services. This is also true in the development of nuclear systems. Therefore, in addition to the development of the global European HPC capacity, **efforts must be made in the nuclear community to have the investigations on nuclear materials recognized as a top priority subject in Europe and guaranteed access to significant computational resources.** This is a foremost condition for advanced modelling to bring the appropriate support to the development and qualification of the materials needed for the ESNII prototypes.

6.6 Costs related to crucial infrastructures build and use

The construction and maintenance of infrastructures and facilities such as MTRs and hot cells is not in the scope of the EERA JPNM activities and capabilities. It is important, however, to emphasise the importance of this point. In particular, if it is out of place to attempt estimates of infrastructure investment costs here, it is useful to remind that they have been detailed in European projects such as ADRIANA,⁸⁷ and vary considerably, with material testing reactors lying in excess of 1000 M€, and hot cells, depending on scope, in the 100-300 M€ range. Typical single fuel irradiation experiments cost 1-2M€, depending on complexity, instrumentation required, and PIE effort. The cost of the transport of irradiated fuel can reach 150 k€, although in a well-planned oriented approach can drop to as little as 15 k€; active structural material specimens can be transported with costs varying between 5-30 k€. The advent of micro sampling techniques (e.g. FIB) for dedicated examination on ultra-small samples can reduce these costs even further.

In addition, Tier-0 HPC resources, which do not serve the whole European scientific and industrial community, have to be renewed every 2-3 years, with construction costs between 200 and 400 M€ and annual running costs around 100 M€.

6.7 Outlook: renewal, sharing and joint programming

A research agenda can only be fulfilled if it is matched by appropriate infrastructure, timely available. Therefore, in the present context, Europe should, through its research organisations located in different MS, and thus with the full commitment of the latter:

- Ensure that a fast neutron flux facility comes available soon (e.g. ESNII prototype);
- Construct at least one new MTR in addition to JHR (i.e. PALLAS, MYRRHA, the latter with the added value for GenIV applications to offer a fast neutron spectrum);
- Renew and possibly extend its hot cell capacity, taking due account of transportation issues;
- Ensure that hot laboratory capabilities, in particular for fuels, do not decrease further;
- Plan major infrastructures judiciously at a pan-European level;
- Ensure appropriate access to high performance computing
- Implement consequent open access and infrastructure sharing initiatives;
- Foster Joint Programming (fusion energy provides an example to follow).

⁸⁷ http://ojs.ujf.cas.cz/~wagner/transmutace/erinda/presentations/05_ADRIANA_ERINDA.pdf.

Here a few relevant considerations follow concerning mainly the last two points, i.e. infrastructure open access and sharing.

Access to and sharing of facilities for nuclear materials exposure, testing and examination, especially controlled zones where radioactive materials are manipulated, **can be problematic** for legal, security, safety and financial reasons, namely:

- **Legal:** Protection of know-how & expertise: there is often reluctance to give full open access as this may reveal details on protected know-how (this attitude is however often inconsistent even within the same organisation);
- **Related to security:** Access to hot cells requires clearance from authorities for security reasons: this takes weeks or months and a significant administrative burden;
- **Related to safety:** Only trained & skilled operators can safely use some equipment, especially in hot labs (manipulators...);
- **Financial:** Availability of specific equipment has a high cost and opening it for access to external users might limit the profitability of facilities.

Notwithstanding these difficulties, **solutions can be envisaged**, for example a scheme of mutual compensation between organisations within a “virtual research centre”, in-kind or in-cash. An in-kind type of compensation is for example by seconding employees from A to B in a stable way. This could happen under specific bilateral but also multilateral agreements concerning non-disclosure of know-how, use of manpower for host purposes, etc.

It should be stressed that mobility scheme of this type provide also a **key motor to drive education and training** of not only researchers, but also of operators, enabling Europe to maintain and actively manage key competences currently dwindling in the nuclear field, for the benefit of all nuclear research centres (see section 7).

In this respect, joint programming is key. For example, to make the most rational use possible of available facilities and join forces in terms of bearing costs, it would be possible to think of the **creation of a permanent joint forum on the planning of irradiation infrastructure use**. This forum should agree upon manufacturing and irradiation needs, as well as design joint campaigns, making use of available facilities, defining criteria for the justification of the use of irradiation facilities outside Europe, expressing views on the rules for financial coverage for the use of nuclear infrastructures, possibly launching tenders and/or using differences between available reactors to explore specific effects, and including, whenever suitable, fundamental studies and the use of ion irradiation. It could also make the best of available space in reactors. For instance, while MTRs operation is extremely expensive and cannot be offered for free, under some conditions it is possible to perform cheap “piggy-back” irradiation experiments, in a concerted framework. Such a forum could be the place within which to coordinate this type of actions, whenever space is available in a device.

It should be clear, however, that **infrastructure sharing and rational planning will be key drivers in the future, but will only be effective within communal joint programming schemes and provided the willingness of sharing and joint programming exists at all levels, including high management level of research centres as well as MS level. This should also be properly fostered by Europe-driven actions.**

7. Education, training and mobility of researchers

Nuclear fission research and competence in Europe may be in danger of decline due to the decrease in the number of new researchers and operators entering the nuclear community, while they are required to replace previous generations. This endangers the possibility of preserving the existing knowledge and skills. Several factors can be identified as origin of this problem, mainly related to the uncertain professional perspectives that the nuclear field currently offers in Europe, as well as in part to the lack of sufficiently funded challenging and attractive projects, such as those offered for example by fusion energy, renewable energy or energy integration. A **suitable and inherently attractive education and training (E&T) programme is therefore required to reduce the risk of a future shortage of nuclear skills and ensure the maintenance of the acquired knowledge and expertise.**

In this respect, it is recognised that **nuclear materials can be a way to attract young researchers to the nuclear field**, thanks to the inherent cross-cutting nature of materials science through several technologies. Nuclear materials courses with a specific focus on nuclear structural and fuel materials with interdisciplinary integration and emphasising both research and industry needs, **may act as catalysers to attract more young**

researchers to work on nuclear energy projects, integrating expertise from other fields into specific nuclear applications. GenIV plays here an important role as it involves highly interesting scientific challenges.

Young researchers and operators should be trained in particular by using both common and specialized facilities that, regardless of size, provide a key engine to drive education and training, enabling Europe to maintain and actively manage currently dwindling key competences in the nuclear field. In this respect, **infrastructure sharing is a key driver, to be made effective within joint programming schemes** (see section 6.7). A homogenous and coordinated network of nuclear facilities and infrastructures, together with a **stable scheme of mobility of young researchers and operators**, is expected to enable wider competences to be built on various aspects of nuclear technology and, by exploiting its inner cross-cutting aspects, also towards other technologies. Some exchange schemes have already been conducted or are planned in the framework of FP7 and H2020 projects (e.g. GENTLE and ENEN+).

To obtain maximum benefit, an actively managed cross-European education and training and mobility (E&T&M) programme with specific focus on nuclear structural and fuel materials should apply outcome-based teaching methods, focusing on **interdisciplinary integration** and emphasizing both **research and industry needs**. It should **extend and complement existing university curricula**, training programmes and schools, without doubling them, while being **recognised by European universities in terms of credits**. To be attractive, it should offer the possibility of **acquiring knowledge and skills related to materials science that can also be useful outside the nuclear field**, as well as the possibility of **hands-on training via access to available experimental facilities**. The ultimate goal here is that an attractive extra-curricular course on materials for advanced nuclear systems of the type described **should be offered periodically to all students and young researchers** that feel interest for this field.

In order to reach this goal, the EERA JPNM considers that a **coordination in terms of education and E&T&M initiatives through nuclear platforms, i.e. SNETP and its pillars, in close collaboration with the European Nuclear Education Network (ENEN), is the right way to go**. ENEN has indeed the aim to encourage E&T related co-operation and integration between institutions and to attract qualified students to participate in special E&T national and international programmes. In collaboration with ENEN and in coordination with SNETP and its pillars, the **EERA JPNM** can then act as the **focal starting point to assure up-to-date and sustainable continuity in the E&T programmes for nuclear structural and fuel materials**, by identifying and providing the high-level experts and research infrastructures which are available for training purposes, as well as suggesting suitable training topics and schemes.

The harmonisation of E&T&M activities between EERA-JPNM, SNETP and its pillars, with the support of ENEN, should also move in the direction of **identifying a series of periodic short training courses, summer schools or workshops that address topics considered important for the whole nuclear energy community, and organising them by making coordinated use of the funds distributed through different Euratom-supported projects for E&T purposes**. The idea here is to reverse the current situation, where each project is delegated the task of identifying suitable E&T activities, to replace it with a cross-platform joint medium term planning of E&T activities, to be funded through the sources that case-by-case are available from projects, topped up with the financial and logistic support that the different organisations involved may offer. Similarly, mobility schemes should be set up, as well.

Depending on the existence of suitable funding schemes, the EERA-JPNM proposes here a potential concept to develop an E&T programme that should offer different learning packages, to give the participants the possibility to acquire specific and needed competences, as described in Table 16.

Table 16: Description of a possible E&T programme on nuclear structural and fuel materials, as envisioned by the EERA JPNM.

| | |
|--------------|---|
| What | <ul style="list-style-type: none"> • A learning package - E&T program devoted solely to nuclear materials (structural materials and fuels) |
| When | <ul style="list-style-type: none"> • Annually or bi-annually, at predetermined dates |
| Participants | <ul style="list-style-type: none"> • Master's students from relevant engineering disciplines • PhD students • Employees from the nuclear industry |
| Class size | <ul style="list-style-type: none"> • Class size is optimally 20 people, max. 30 people |
| Where | <ul style="list-style-type: none"> • Suitable locations can be identified after sending out respective explanations and a questionnaire to all known institutions asking for their availability • Sessions focusing on computational exercises, lectures and in-class problem solving are not particularly location-dependent and can be held alongside other workshops, conferences etc. of the nuclear material community • Suitable experimental facilities must offer at least partial hands-on experience to students |

| | |
|-------------------------------|--|
| Length | <ul style="list-style-type: none"> • 4 weeks in total in suitable institutions, consisting of 2-4 individual sessions (2+1+1 weeks or 2+2 weeks or 1+1+1+1 weeks) • 1-2 weeks of optional, preparatory online learning in advance to bring everybody up to the same level • Participation during the entire year is necessary to meet the goal of high-level competence |
| Teaching methods | <ul style="list-style-type: none"> • Hands-on experimental assignments • Hands-on computational exercises • Learning through in-class and at-home comprehensive problem solving • Lectures leading in assignments and assisting in their solving • Field trips to the industry and research facilities |
| Topics | <ul style="list-style-type: none"> • Suitable topics for the 4-week period should be defined by the trends for future research needs, the needs of students and the competence of teachers willing to participate |
| Teachers | <ul style="list-style-type: none"> • Outstanding researchers and experts from universities and the industry who are interested in teaching and applying effective teaching methods to achieve the intended learning outcomes |
| Teaching support | <ul style="list-style-type: none"> • A short course to motivate applying successful teaching methods • Guidelines for desirable lecture composition and assignment design |
| Suggested curriculum features | <ul style="list-style-type: none"> • Topics per session limited to a small number so that sufficient time is given to students to listen, understand and apply new knowledge • A high problem-solving-to-lecture ratio should be applied • Teacher-assisted independent work in small teams throughout all sessions • All assignments related to topical research interests and industry interests |
| Management aspect | <ul style="list-style-type: none"> • Fewer topics per week means fewer teachers per week which results into smoother time management and potentially reduced costs • Annual course on predetermined dates simplifies planning and availability both for teachers and students |

8. Industry and regulators involvement

The EERA-JPNM is a public research organisation platform that does not deal with the actual design, licensing, construction and operation of GenIV prototypes where, ultimately, materials are going to be actually used and therefore fully qualified, based on return of experience, in the proper operational environment. This is the task of industrial initiatives, specifically ESNII, and in particular of the industrial partners involved. In terms of technology readiness level (TRL) the EERA-JPNM mainly works between TRL 2 and 5, i.e. just up to the level where the material becomes industrially usable, by having been developed and qualified to the pre-normative level.

It is however essential to have a close link to the industrial application. In particular the goals and the needs of the reactor designers must be very clear in order to support the processes of licensing and construction of advanced nuclear systems, for which return of experience is limited or even non-existent. Therefore, **the involvement of industrial reactor designers and constructors in the definition of the research agenda priorities of the EERA JPNM is essential.** These are almost all involved in ESNII, thus ESNII acts as industrial counterpart for the EERA JPNM. It is equally essential that **the results of the research of the EERA JPNM are appropriately and efficiently transferred to the industrial reactor designers and constructors,** as a minimum in the form of data/knowledge, and possibly through suitably updated design codes. This problem is also connected with the problem of data management, addressed in section 2.4.

In this framework, although the interaction with regulators is mainly the duty of the system designers, because of the need to refer to specific designs and service conditions, it is argued that **the involvement of TSOs and regulators to follow the procedures used for materials qualification and possibly guide them is likely to further accelerate the licensability of nuclear components.**

The connection with industry is also essential in the process of development of new materials, especially when innovative fabrication routes need to be explored that should, eventually, be upscaled to industrial production. This upscaling is not necessarily simple, because it is not guaranteed that a certain type of material can be efficiently and affordably produced at larger production scale outside the laboratory where it has been developed. For example, currently no large scale industrial producer of ODS steels exists in Europe, despite the fact that several laboratories can do that, in small or even very small batches. Industrial production may require adaptation or even complete changes in the fabrication, processing and treatment of

the materials. Thus it is very **important that work hand-in-hand with the industry starts as early as possible along the materials development route**, in order to take into account industrial production upscaling as a criterion. The industrial counterpart is, in this case, represented by materials manufacturers, for example steel-makers.

The industrial involvement in the EERA JPNM activities is achieved in two ways:

1. **Industries may join the EERA JPNM as associate members.** Currently a few industries, both reactor designers/constructors, and materials manufacturers (steel-makers), participate actively in the EERA JPNM projects, also with in-kind contributions, as partners.

2. **Industrial representatives** that prefer not to commit themselves to active participation in the EERA JPNM projects, but are ready to provide guidelines and feedback to the EERA JPNM activities and to have partial access to the results obtained (following case by case IPR rules and based on the signature of a non-disclosure agreement) **are invited to join the stake-holders' group (SHG).**

SHG members exert their function by participating in bespoke workshops and meetings, as well as by contributing to task forces that are meant to design research plans and strategies for specific issues that have industrial implications. The wish for the future is that also TSOs and regulators may join the SHG of the EERA-JPNM.

9. International cooperation

International cooperation can provide a real boost to progress towards innovation in the nuclear energy field, which is much needed along the path leading to the deployment of GenIV systems.

However, for international cooperation to be effective, **appropriate instruments and incentives need to be put in place in Europe towards other countries**, given that those that exist right now are clearly insufficient, not sufficiently efficient, or in the best case not sufficiently attractive (and known) to researchers.

Areas where effective international cooperation could determine an important boost in the development and qualification of materials for advanced nuclear systems, provided the right instruments are offered, are for example:

- **Optimized/harmonized use of infrastructures**, making unique facilities transnationally accessible and also, importantly, adequately planning their use to avoid duplications and redundancies, while making the best use possible of the specificities of each particular infrastructure: this should be the rational approach to make materials qualification complete and affordable.
- **Harmonization of procedures and methodologies** to test and characterize materials, especially innovative materials in specific environments, including protocols to perform microstructural examination with advanced techniques and to analyse the results. In several cases, completely new tests need to be designed and standardized. Moreover the design codes RCC-MRx and ASME BVP Section III share the same goals so also in this area there is of mutual interest to share and compare methodologies for design rules and design curves.
- **Data collection and sharing**, through suitable databases that should be eventually made available for reactor designers (industry and not only) and regulators: the more complete and extensive the databases, the safer the corresponding design rules and the more conducive the action of the regulators. Importantly, data collection and sharing makes sense provided that tests are homogeneous as to procedures used, so this point links strongly with the above one.
- **Synergy on modelling**: modern modelling approaches at different scales inherently require inter-disciplinarity, computer resources, theoretical developments and, crucially, extensive experimental effort for the characterization of materials at all scales after exposure to a variety of conditions. Structured international coordination and exchange can only be beneficial. The three points above (optimal use of infrastructures, harmonization of test and characterization procedures, and data collection and sharing) clearly enter here too. In addition, there is a need to make modelling approaches more compatible and complementary with each other, in order to better focus the development towards platforms of linkable codes and models.

Enablers can be identified in terms of international cooperation schemes. For example expert **committees could be enabled to work on the issues listed here below.**

- **Harmonisation of test and characterisation methodologies and procedures**, as pre-normative step to standardization by bespoke bodies (ASME, CEN, ISO, ...): support to round-robin exercises addressing non-standardized tests; best practices for microstructural characterization technique application and data analysis, with exercises of inter-comparison between laboratories for specific microstructural characterization techniques (TEM, APT, SANS, PAS, ...).

- **Identification of unified international databases of reference** for the collection of materials testing data; establishment and distribution of relevant data templates compatible with selected reference database(s); establishment of rules concerning protection and disclosure of data collected in databases (e.g. 10 years embargo on proprietary data); encouragement to upload data: this is always the bottleneck in the case of databases, thus grants should be accorded to support financially data seekers and collectors.
- **Overview the use of major research facilities worldwide** (mainly materials test reactors, but also facilities for the exposure of materials to high temperature and coolants), based on existing lists and maps; optimisation of complementary use of these facilities, possibly driving similar facilities to non-overlapping uses; design of large joint experimental programmes of (possibly) cross-cutting interest, using available large facilities in a coordinated way (this may specifically apply in the case of modelling-oriented experiments).
- **Harmonization in the development of computer simulation materials models** for better mutual complementarity and compatibility; identification of gaps; modelling data collections (according to criteria similar to the above ones applying to testing and characterisation).

The role of international organisations such as NEA/OECD or IAEA is pivotal in this respect, to facilitate cooperation on nuclear energy at global level. For this reason, EERA signed a MoU with NEA/OECD, that is intended exactly to bring the JPNM to a higher level of international visibility, by actively participating in, and providing expertise to, NEA initiatives such as working parties, expert groups, task forces, and so on. Specifically, an opportunity for effective international cooperation in which the EERA JPNM is fully involved is currently offered by the NEA Nuclear Innovation 2050 Initiative.⁸⁸

In concrete terms, **a more effective interaction with the GenIV International Forum (GIF⁸⁹) is advocated.** GIF is a cooperative international endeavour based on a Charter that was signed first in 2001 by a number of countries, that has now grown to 14. While some countries are “dormant” members, especially active are United States, South Korea, France, Switzerland, and Australia that recently joined, as well as, to a lesser extent, Japan and also China. The Russian Federation is also member of the GIF. The goal is here to share data of relevance for GenIV systems. Euratom is also a signatory of the GIF Charter, even though only a minority of the EU countries are involved. Thus **the connection with GIF should occur mainly through Euratom**, with the pivotal role of JRC as interface. The EERA JPNM has for the moment only informal contacts with the GIF project devoted to materials for VHTR, in whose framework some activities of cross-cutting interest through several GenIV systems are included. Stronger connections can only be established within a legal support provided by Euratom.

In terms of specific non-European countries, benefit is quite obviously expected in particular through collaboration with those involved in the GIF, so United States, South Korea, Australia, Japan, China and Russian Federation, because all these countries have specific activities on GenIV materials. Data sharing is the essence of the GIF, but most of these countries should also be interested in harmonizing testing and characterization procedures, without which data sharing becomes less meaningful. Collaboration with the United States on several fronts, from materials qualification to modelling and development of new materials is relatively easy and instruments such as I-NERI⁹⁰ are in place, although they are not in practice very attractive, especially on the European side. Collaboration with South Korea on issues related with creep design rules for austenitic and F/M steels and compatibility of these materials with HLM is ongoing through Euratom projects (FP7/MATTER, H2020/GEMMA), in which Korean institutions participate as partners. In principle a similar type of collaboration could be extended to US partners directly involved in the development of the ASME code. More difficult or less exploited is the collaboration with the other countries. Modelling could be a good ground for collaboration with Australia. Russian Federation, China and potentially also Japan are important partners to be considered because of the unique opportunity they may offer to expose materials to high dose in fast spectrum reactors, such as BOR60 in Russia, where for example experiments were performed in the framework of the FP7/GETMAT project. However, efficiency can only be guaranteed if (whenever needed) **suitable agreements are signed** (e.g. between Euratom and EU member states and other countries) and **appropriate funding is provided**, not only to cover meeting and travel expenses of experts, but also for the work performed in preparation of them. Ideally, **funded schemes for mobility of researchers** should be set up in support of the four enablers listed above.

⁸⁸ <https://www.oecd-nea.org/ndd/ni2050/>.

⁸⁹ https://www.gen-4.org/gif/jcms/c_9260/public.

⁹⁰ E.g. http://cordis.europa.eu/fp7/euratom-fission/docs/06-haas-ws4_en.pdf

10. Resources

It is challenging to foresee correctly the costs associated with research that is not meant to lead *directly* to the development of a defined technology and its release on the market, or to the construction of a system, but rather to provide the suitable and continuous R&D&I background to *enable* such technology to be developed and released, as is the case for nuclear materials for advanced nuclear systems. However, based on past experience and also on estimates coming from the deployment of the research agenda of the EERA JPNM, **excluding the cost of the construction and maintenance of the needed infrastructures and facilities**, which are taken for granted, one can assess that:

- In order to adequately address the most important issues, strictly related with the needs of the ESNII system prototypes and their follow up (so, fast GenIV reactors, see Table 14), with very limited industrial involvement and also limited activities devoted to perspective innovative materials, **excluding the cost of irradiation campaigns, subsequent transport and use of hot cells, a minimum of about 12-15 M€ per year should be ensured.**
- If *all costs* of irradiation qualification campaigns, transport and hot cells are included, for proper materials qualification (within the limits allowed by the facilities that are available), then **the resources should range between 20-25 M€ per year.** This amount should also enable adequate support for education, training and mobility of young researchers.
- The inclusion of significant industrial involvement and extensive activities also on materials currently at low or very low TRL, extended to systems beyond those included in ESNII (i.e. VHTR, MSR, SCWR,...), would about double the costs, leading to the upper bound estimate of **50 M€ per year, as expressed in the Integrated Roadmap of the SET-plan.**⁹¹

These resources clearly exceed by about an order of magnitude the possibilities of funding offered by Euratom. Therefore, not only the *involvement*, but also the *commitment* of the Member States (MS) is crucial. MS funding for these activities already exists now, in particular stable salaries for researchers and the possibility of opening new positions in the field are guaranteed through MS provisions and support to research centres and universities, meaning that there is already very significant involvement. However, **there is currently no coordinated commitment to earmarked MS funding**, similarly to what happens for instance in the case of fusion energy. Fusion energy receives support under different forms, including through joint undertakings (Fusion for Energy, F4E) and international agreements (ITER, Broader Approach). One form of support is through the instrument called European Joint Programme (EUROfusion). This instrument appears to be especially suitable to enable the joint implementation of a roadmap, using, amongst other tools, internal calls for projects, including schemes of E&T&M, with a clear commitment of member states and the support of Euratom. Of course, one of the main reasons for setting up such a funding scheme for fusion energy is that the support to this research is widespread through essentially all EU members states. Yet, nuclear materials at large are actually a subject that, for different reasons and purposes, can find widespread support through several member states, by virtue of its cross-cutting nature. The present SRA is a good starting point and shows the existence of an already established and thriving European nuclear materials research community, partly under the umbrella of the EERA JPNM but partly also under the umbrella of SNETP pillars, NUGENIA *in primis*. It is therefore believed that such an instrument, or equivalent **co-fund instrument, could be suitable also for the established nuclear materials community.** However it is also emphasised that, without suitable support for projects specifically aimed at designing and constructing GenIV prototypes, such as those considered in ESNII, there will be no serious driving force to sustain these activities, beyond fundamental research motivations.

Figure 8 shows the approximate subdivision of resources through EERA JPNM activities, based on both the estimated costs of EERA JPNM pilot projects from the recent past, as well as from the actual cost of research activities in the framework of European and national or institutional projects.

⁹¹ Heading 5 of Part II: “Supporting Safe Operation of Nuclear Systems and Development of Sustainable Solutions for the Management of Radioactive Waste”. Action: “Qualify nuclear materials for operation under Gen IV conditions and develop innovative materials to improve plant safety and efficiency”. See also Section 1.6.2.

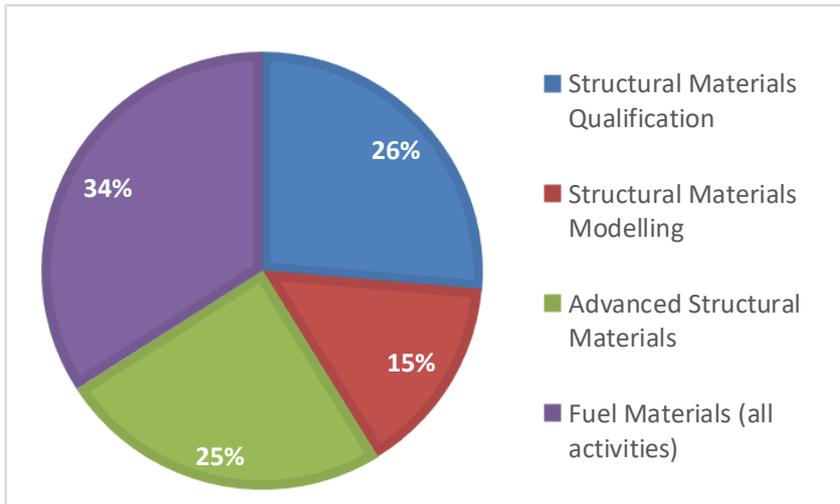


Figure 8: Indicative estimated subdivision of funds between research activities, based on past experience, assuming all activities are sufficiently funded.

11. Summary and recommendations

This SRA identifies the research lines to be pursued in order to ensure that suitable structural and fuel materials are available for the design, licensing, construction and safe long-term operation of GenIV systems, with emphasis on the fast neutron spectrum systems considered in ESNII, namely SFR, HLM-cooled systems (ADS and LFR), and GFR. These create more fuel than they burn and operate at high temperature using passive safety systems, so increasing enormously the sustainable use of resources and guaranteeing safe energy production for centuries, significantly reducing waste production.

The materials considered here cover as priority the needs of the ESNII prototypes and demonstrators, but attention is put also on materials solutions that are intended for FOAK and commercial GenIV systems, in which higher energy efficiency and longer burnups than in the prototypes are targeted. Importantly, the content of this SRA is fully consistent with other relevant strategic documents and roadmaps compiled by other platforms and in other frameworks.

The research activities are organized in blocks that result from the application, for structural and fuel materials, of a well-established materials science approach based on the combination of three classes of activities: (1) materials testing and characterization for full qualification and definition of design rules in a pre-normative spirit; (2) development of mechanistic and physical models in support of materials behaviour correlations used in design rules and improvement of materials properties; (3) development of advanced materials through experimental screening of solutions, also assisted by models rooted in the understanding of the physical processes that govern materials behaviour.

Effort has been devoted to identify issues addressed in a GenIV materials research framework that are also of common interest for other nuclear and also non-nuclear energy technologies. This is done with a view to optimising the use of available resources, whenever possible, by joining forces with other research communities.

The document addresses also issues that are necessary corollaries to the proposed research activities, namely: infrastructures needed, need for education and training and mobility schemes, industry and regulators involvement, importance of international cooperation, estimate of resources required.

Concerning the latter issues, and besides the obvious need of adequate financial resources in order to address the research problems outlined in this document, **a few recommendations emerge** that this document intends to bring to the attention of stake-holders, particularly research managers and decision-makers:

R1: Data from materials property measurements after exposure are the essential ingredient for robust design curves and rules. Plenty of data were produced in the past that are de facto unusable either because covered by confidentiality or because not properly stored. Correct data management to guarantee their availability for

future re-assessment is therefore essential and should be encouraged and fostered. In particular, **financially supported policies to foster data sharing and to encourage old data disclosure should be implemented.**

R2: Some infrastructures are absolutely essential to enable the correct qualification of nuclear materials, not only irradiation facilities, but also suitable hot cells where active materials can be safely handled and tested, nuclearized characterization techniques, loops and pools for compatibility experiments, etc. They are also crucial for education and training of young researchers and operators. These infrastructures are costly to build and maintain. Other research facilities are, on the other hand, more common and sometimes redundant. **A rational and harmonised, trans-national management of infrastructures in Europe, including schemes for facility sharing, would be highly desirable and, at the end of the day, beneficial for all.**

R3: International cooperation with non-European countries where research on nuclear materials is pursued can be very valuable for Europe. Quite clearly, the goals of this cooperation are the same as in the case of internal European cooperation, namely coordinate activities, share data, have access to infrastructures. Currently, however, **the instruments available in Europe for international cooperation are not sufficiently attractive to motivate researchers, so more efforts should be made to improve their attractiveness and ease of application.**

R4: The nuclear materials research community in Europe is currently strongly integrated and engaged in thriving collaboration, in a bottom-up sense, much more than at the level of instruments offered to make this integration efficient and functional in a top-down sense. This SRA is largely the result of matching bottom-up research proposals with top-down strategies. The appropriate instrument to allow this community to deliver according to the goals should provide the conditions to implement the agreed upon research agenda and to set up suitable E&T&M schemes allowing knowledge, data, and facility sharing. Since the financial support of Euratom will never be sufficient, earmarked funding from the MS, dedicated to support integrated research on nuclear materials, is crucial. In this sense, **co-fund instruments, like a European Joint Programme, seem to be the most suitable ones.**

These recommendations are clearly based on the willingness of pursuing a policy of increased integration rather than of isolation, at all levels: research organisations, EU Member States, and European Commission. Besides the amount of resources that can be reasonably allocated, this requires finding a difficult equilibrium between the need to make the best use possible of the limited resources available in a framework, nuclear energy, where support is politically not simple to obtain, and the legitimate ambition to preserve everyone's assets, in a context of healthy competition.