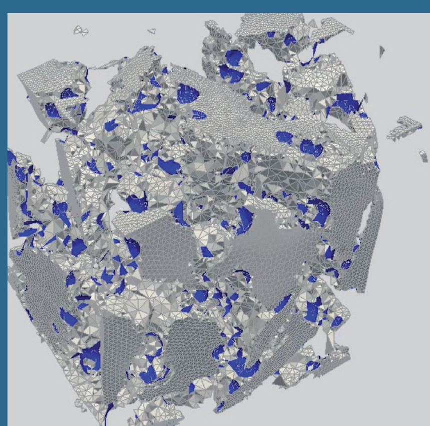
MATERIALS FOR SUSTAINABLE NUCLEAR ENERGY

The Strategic Research Agenda (SRA) of the Joint Programme on Nuclear Materials (JPNM) of the European Energy Research Alliance (EERA)



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Joint Programme on Nuclear Materials of the European Energy Research Alliance
Coordinating sustainable nuclear materials research for a low carbon Europe

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Summary

List of abbreviations

ADS	Accelerator-Driven System
AFA	Alumina Forming Austenitic (steels)
AI/ML	Artificial Intelligence / Machine Learning
AMPEA	Advanced Materials and Processes for Energy Applications (EERA JP)
ATF	Accident Tolerant Fuel
APT	Atom Probe Tomography
BU	Burn-Up
C/M	Carbide/Metal
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
ESTEP	European Steel Technology Platform
EMIRI	Energy Materials Industrial Research Initiative
EUMAT	European Technology Platform for Advanced Engineering Materials and Technologies
ETIP	European Technology and Innovation Platform
ETP	European Technology Platform
E&T(&M)	Education and Training (and Mobility)
EU	European Union
EURATOM	European Atomic Energy Community
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
HPC	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
IAEA	International Atomic Energy Agency
IEA	International Energy Agency
IPR	Intellectual Property Rights
IR	Integrated Roadmap (SET-plan)
JPNM	Joint Programme on Nuclear Materials
JOG	Joint Oxyde-Gaine (French for oxide-clad joint)
KEMS	Knudsen Effusion Mass Spectrometry
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)
MSR	Molten Salt Reactor
MS	Member State
MTR	Material Testing Reactor
M2F	Expert Group on Multiscale Modelling of Fuels of the WPMM (OECD/NEA)
NC2I	Nuclear Cogeneration Industrial Initiative

NDE	Non-Destructive Examination
NEA	Nuclear Energy Agency
N/M	Nitride/Metal
NPP	Nuclear Power Plants
NUGENIA	NUclear GENII/III Association
ODS	Oxide Dispersion Strengthened (or Strengthening)
OECD	Organisation for Economic Co-operation and Development
O/M	Oxide/Metal
PAS	Positron Annihilation Spectroscopy
PIE	Post Irradiation Examination
PLD	Pulsed Laser Deposition
PyC	Pyrolitic Carbon
ROG	Réaction Oxyde-Gaine (French for oxide-clad reaction)
R&I	Research and Innovation
R&D	Research and Development
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
TSO	Technical and Scientific support Organizations
(V)HTR	(Very) High Temperature Reactor
WPMM	Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems (OECD/NEA)
XRD	X Ray Diffraction
2DS	2 Degree Scenario

PREFACE

Nuclear energy in the European Union and its associate countries is experiencing a difficult transient. However it is obvious that, worldwide, nuclear energy will contribute substantially to the low-carbon energy mix that is expected (and hoped) to limit global warming in the coming decades. In contrast, in the EU only three member states are currently building new nuclear power plants: France (1), Finland (1), and Slovakia (2); and only a few more are planning or considering new builds, mainly in Central or Eastern Europe, as well as the UK. So, in the period to 2030 more nuclear capacity will be lost, due to the closure of old reactors, than gained from new ones. Accordingly, the Nuclear Illustrative Program (PINC) issued in 2017 by Euratom forecast a decline in EU nuclear capacity to 2025 followed by (hopefully) a levelling out to 2050. Global energy in general, and nuclear energy in particular, require a strategic and technical long-term vision for a safe and decarbonized energy supply, which seems to have been lost these last years due to political and populistic debates.

The rapid and welcome development of renewable energy sources poses new challenges for the global energy system and particularly for the security of supply. The need of flexible power capacity to complement fluctuating sources is one of these major challenges. Nuclear is able to regulate fluxes and is one of the few available solutions, as well as fossil fuels during the transition period, but nuclear currently lacks market signals to make its production economically attractive, specifically suffering from adverse investment environment. Here, not only economic, but also technological issues are at stake, largely connected with the capability of materials to withstand operational conditions for which the existing plants were not conceived. Facing this problem requires, among other political and economic measures, innovation in the field of materials for power plants, in terms of wider performance and lower cost.

Nuclear energy in particular requires innovation in a broader sense. While the level of safety of nuclear installations in Europe is probably one of the highest in the world, it can be further improved with better materials. In addition, there is a need for better flexibility and adaptability in a wider spectrum of applications, and to pursue the ultimate goal of sustainability by improving efficiency, reducing waste volumes and hazard and, through fuel breeding and recycling, making available resources sufficient to fuel power plants for the next centuries and, why not, millennia.

This Strategic Research Agenda, elaborated by the EERA Joint Programme on Nuclear Materials largely during my mandate as chairman of the Executive Committee of EERA, is a small, but still important, contribution to solve the above challenges. It is therefore with satisfaction that I commend the work done by the management board of this joint programme to let this document see the light and recommend its reading and adoption by the involved policy and decision makers, research managers, and researchers at large.

Hervé BERNARD

CEA, former EERA Chair

EXECUTIVE SUMMARY

With a yearly production in excess of 850 TWh_e, nuclear energy is the single largest source of low-carbon electricity in the EU. Consistent with this, the 2 degree IEA scenario concludes that 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, in the energy transition** from fossil fuels. Yet, three main open issues remain: (1) sustainable and responsible use of resources; (2) accident risk; and (3) long-lived nuclear waste.

The sustainability of nuclear energy will be ensured by deploying Generation IV (GenIV) systems. These can (i) produce more fuel than they consume, guaranteeing low-carbon energy production for centuries through recycling, without additional mining, in a circular economy; (ii) offer ~50% higher thermal efficiency and increased standards of passive safety than current reactors, thereby becoming both societally and economically attractive; and (iii) reduce significantly the volume and radiotoxicity (decay time < 1000 years) of nuclear waste. However, materials will be exposed to high levels of temperature and irradiation, with some in contact with potentially aggressive non-aqueous coolants, targeting a 60 year operation reactor design. Thus, the **development, screening and qualification of suitably performing and affordable materials are crucial** to make GenIV reactors an industrial and commercial reality, with positive impacts on economy, safety, waste, and thus 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 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 the European Sustainable Nuclear Industrial Initiative (ESNII), namely sodium-cooled fast reactor (SFR), heavy liquid metal (HLM) cooled systems (accelerator driven systems –ADS- and lead-cooled fast reactor –LFR-), and gas-cooled fast reactor (GFR). However, links with other GenIV systems, namely (very) high temperature reactor ((V) HTR), supercritical water reactor (SCWR), and molten salt reactor (MSR) have been established. Effort has been devoted to identify **GenIV materials' research issues that are of interest for other nuclear and also non-nuclear energy technologies**, as well. This was done with a view to optimise the use of available facilities, knowledge and human/financial resources, whenever possible, by joining forces with other research communities. Importantly, the content of **this SRA is consistent with relevant strategic documents and roadmaps that have been compiled by linked organisations and platforms** (SETPlan integrated roadmap, OECD/NEA technology roadmap, SNETP SRIA, …), of which it is an **updated extension for what concerns materials for sustainable nuclear**.

Three **Grand Challenges** have been identified, namely: (i) Elaborate design correlations, assessment and test procedures for the structural and fuel materials that have been selected for the demonstrators under the service conditions expected; (ii) Develop physical models coupled to advanced microstructural characterization to achieve high-level understanding and predictive capability; (iii) Develop innovative materials solutions and fabrication processes of industrial application to achieve superior materials properties, to increase safety and improve efficiency and economy.

Based on the current status of the research in the field of nuclear material science and on the needs of the industrial stake-holders, and consistent with the Grand Challenges, this document identifies the **categories of materials of interest**, the **research approaches, tools and instruments**, and especially the specific **scientific and technical gaps and objectives**, to enable the **design, licensing and construction** of GenIV system **demonstrators**, as well as to open the way to their **longer term commercial deployment**. The timeframe is defined by the actual plans of construction of the demonstrators, which depend largely on political decisions and support, as well as on the economic contingency. The document also addresses the necessary corollaries to the proposed research activities, namely: infrastructure needs, education and training, mobility schemes, industry and regulators involvement, importance of international cooperation, …

The research activities are organized in blocks that result from the application, for structural and fuel materials, of a **well-established materials science approach**, which is based on the combination of three classes of activities: (1) **materials testing** for full **qualification** in environment 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; and (3) **development of advanced materials solutions** through experimental screening of solutions, assisted by models that are rooted in understanding of the physical processes that govern materials behaviour. Considerations of the circular economy and advanced manufacturing methods based on automation of procedures, as well as **materials monitoring**, also enter this third class of activities. In all cases, **data management** is a crucial issue to ensure data collection, storage, interoperability, and re-usability.

The **materials of interest** for all **demonstrators**, on which the qualification and pre-normative research effort is **mainly focused**, are **austenitic stainless steels** for structural functions (Ni-based alloys for out-of-core applications; in some cases ferritic/martensitic –F/M- steels) and **MOX** as fuel, with **ceramics** (mainly SiC/SiC) and UO_2 as fuel for the high temperature operation of e.g. the GFR demonstrator. **Surface protection** may be necessary already for demonstrators to provide sufficient compatibility with coolants. Improvements of safety, performance and economy in future prototypes and then commercial reactors advises the exploration of **improved austenitic steels**, **advanced F/M steels**, **refractory alloys** (V and Mo), **oxide dispersion strengthening**, **advanced surface protection methods** and, for the farther future, prospective materials such as **high entropy alloys** or **MAX phases**. Reduction of waste and increased safety requires the development of **minor actinide bearing fuel** from multirecycling, as well as advanced fuel forms such as **uranium nitrides or carbides**.

For **structural materials**, the requirement of **60 years design lifetime** for non-replaceable components is in perspective the most demanding requirement, which includes under its umbrella several issues related with the reasonable prediction of long-term degradation processes: **high temperature processes** (creep, fatigue, thermal ageing), **compatibility with** –especially- **heavy liquid metal and helium coolants**, and effects of **low flux prolonged irradiation**, with emphasis on welded components in all cases. In terms of testing, there is a need for standardization, especially for sub-size and miniature specimens. The modelling, supported by microstructural characterization, has as its main objective the development of suitable **microstructure evolution models** to be used as **input to models for the mechanical behaviour** under irradiation and at high temperature, eventually linking with **fracture mechanics**. Specific developments are required for **coolant compatibility models**, as well as for **models in support of the use of charged particle irradiation** for the screening of new materials solutions, such as those listed above.

Concerning **fuel materials**, the properties and processes that govern its behaviour in pile, on which research effort is focused, are: **margin to melting** (establishment of phase diagrams and evolution of thermal properties), atomic **transport properties** and ensuing microstructural evolution, **fission product** (non-gaseous) and **helium** (gas) behaviour and transport, **mechanical properties** (their evolution, subsequent fragmentation and cracking, fuel-cladding mechanical interaction), and **compatibility with cladding and coolant** (internal cladding corrosion, chemical interactions especially in case of severe accident). These are all addressed from both an experimental and a modelling perspective.

Several **materials issues** identified for GenIV demonstrators are **common to other nuclear fission technologies**, namely **GenII/III reactors and (V) HTR**, especially concerning the integrity of structural materials (qualification of welds, procedures for small specimens, modelling and use of charged particle irradiation), fuel (oxides, innovative fuels and synthesis routes, fuel performance codes, modelling, materials for accident tolerant fuel), innovative materials solutions and manufacturing. Most materials and issues related to high temperature operation are common between GFR and (V) HTR.

Important **commonalities exist also with fusion materials**, namely all activities related with F/M steels (design rules, low temperature radiation embrittlement, high temperature behaviour, welding, compatibility with heavy liquid metals, advanced F/M steels), materials screening methods, ceramic coatings, mechanistic and physical modelling of irradiation effects and compatibility with HLM.

More in general, the development and qualification of materials resistant to high temperature and aggressive environment, as well as an approach based on testing and characterization, modelling and development of new materials solutions is **common to several other energy technologies**, such as concentrated solar power, geothermal, bioenergy, fuel cells and hydrogen.

Several classes of facilities and infrastructures are necessary for the qualification of nuclear materials in general, and specifically for GenIV systems. The specificity of nuclear energy makes **irradiation facilities essential for the qualification of nuclear materials**. Ideally GenIV materials should be qualified by exposure to fast neutrons up to high dose, however at the moment **Europe is totally dependent on non-European countries to have access to fast neutron facilities**. Working **materials testing reactors** (MTR) with dominantly thermal spectrum, equipped with relevant hot cell facilities, exist **in Europe**, but there are **not many** of them. Moreover, **hot cells and shielded facilities** where active materials can be handled and tested are costly infrastructures to maintain and, for this reason, are also **limited in number** and often becoming **obsolescent**, despite the construction of new facilities in some countries: this limits severely the number of tests, measurements and examinations that can be performed on neutron irradiated materials. **Transport** of active materials, especially fuel, is also becoming problematic, largely because of the **lack of harmonisation of regulations** throughout Europe. For all these reasons, **charged particle irradiation** facilities become crucial to provide data that, although unsuitable for qualification, can be exploited for **screening and modelling purposes**. For the latter, access to suitable **computational resources** becomes also important, though there is currently **no dedicated resource for fission** material studies. Overall, a number of **actions are needed** in order to make sure that sufficient facilities and infrastructures are available in Europe for nuclear materials research. Amongst them, a better **coordinated use and development** of infrastructures throughout Europe is desirable, based on a principle of complementarity, rather than competition, including **common rules for the access to facilities and infrastructures.**

Strongly linked to use and sharing of infrastructures and facilities is the problem of education and training to their use for nuclear materials research. A **suitable and inherently attractive education and training (E&T) programme** is needed to reduce the risk of a future shortage of nuclear skills and ensure the maintenance of the acquired knowledge and expertise. **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. This fact should be better exploited.

Finally, a **close link** of nuclear materials research **to the industrial application is essential**. In particular the goals and 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 the amount of experience is limited. The involvement of **technical and scientific support organizations (TSOs) and regulators**, to follow the procedures used for materials qualification and to possibly guide them, is also desirable, though difficult, and is likely to further accelerate the licensability of nuclear components. The connection with **materials' manufacturers** is essential as well in the process of development of new materials' solutions in view of industrial upscaling, especially when innovative fabrication routes are explored. For these reasons, explicit efforts are made on the EERA JPNM side to facilitate industrial involvement, e.g. through participation in **task forces**.

International cooperation is currently **not sufficiently encouraged** by the means that are offered. However, international cooperation is **essential** for the optimized/harmonized use of infrastructures, harmonization of testing procedures and methodologies for interlaboratory data consistency, data collection and sharing and also synergy on modelling. The role of **international organisations** (IAEA, NEA, …) to facilitate this is key, in particular to enable better connection with the GIF.

Besides the obvious need of adequate financial resources in order to address the research problems outlined in this SRA, as well as the necessary corollaries, **four recommendations** emerge that this document is intended to bring to the attention of stake-holders, particularly decision-makers:

R1: **Data** from materials property measurements after exposure to relevant conditions are the essential ingredient for robust design curves and rules. Plenty of data were produced in the past that are now de facto unusable; this is either because they are covered by confidentiality or because they were not properly archived. Correct data management to guarantee availability for future re-assessment is therefore essential and should be encouraged and fostered. In particular, **financially supported policies to foster data sharing and 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, **pan-European management of infrastructures,** based on **joint programming,** including **trans-national infrastructure renewal planning and a scheme for facility sharing and exploitation**, 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 coordination of activities, sharing of data, and access to infrastructures. Currently, however, **the instruments available in Europe for international cooperation are not sufficiently attractive** to motivate significant cooperation with non-EU researchers. Efforts should be made to improve their attractiveness and ease of access. International **organisations** such as OECD.NEA, IAEA, but also Euratom and JRC for the connection with GIF, have here a crucial role.

R4: The **nuclear materials research community in Europe is currently strongly integrated** and engaged in thriving collaboration, in a bottom-up sense. This is in contrast with the inadequacy of the top-down instruments offered to make this integration efficient and functional. 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 SRA goals should provide the conditions to implement the agreed research agenda and to set up suitable E&T&M schemes that allow 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, a **co-fund instrument, such as a European Joint Programme, seems to be most suitable**.

These recommendations are clearly based on the willingness to pursue a policy of increased integration rather than of isolation at all levels: research organisations, EU Member States, and European Commission. Given 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 the finding of a difficult equilibrium between the need to make the best use possible of the limited resources available, in a framework of nuclear energy where support is politically difficult to obtain, and the legitimate ambition, in a context of healthy competition, to preserve each stakeholders assets.

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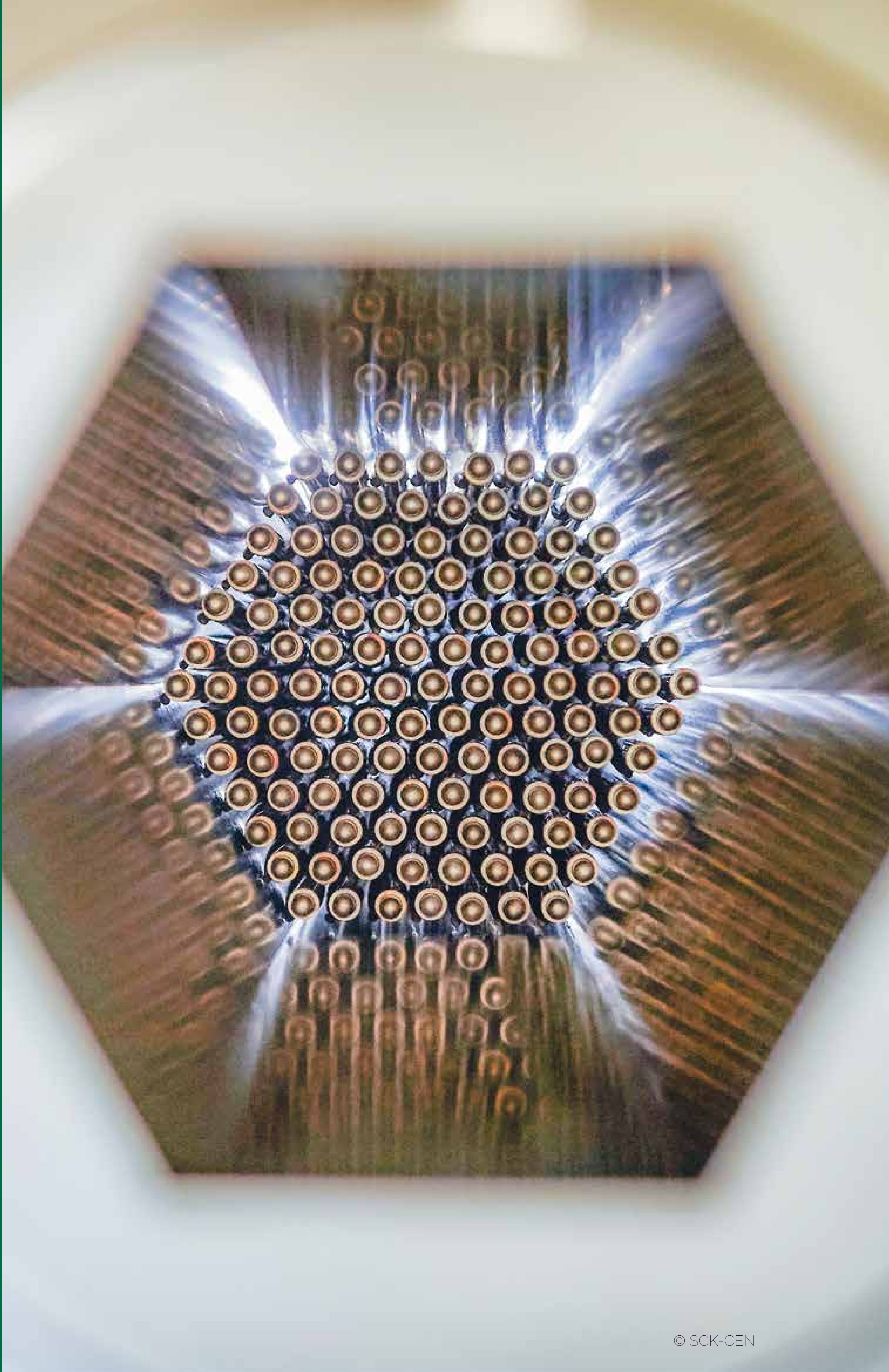
INTRODUCTION



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 electricity in the European Union.[1] This corresponds to more than ¼ of the electricity in Europe, and contributes to guaranteeing a secure and reliable base-load supply. Consistent with this, in the 2 degree scenario (2DS) of the International Energy Agency (IEA), it is considered that 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 the energy transition. However, three main open issues remain: (1) sustainable and responsible use of resources; (2) accident risk; and (3) long-lived nuclear waste.

Current nuclear systems are insufficiently sustainable: just less than 1% of the fuel's 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 deploying 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 closed fuel cycle allows the energy extracted from the available uranium resources to be maximised. Fuels irradiated by fast neutrons generate as much Pu from the ^{238}U by neutron capture as the ^{235}U is consumed by fission. The reactor cores can be thus optimized, pushing the burnup to high values, i.e. letting the fuel remain for longer in the reactor, to produce more Pu than they consume (breeder reactors). This can be then extracted and used for refuelling, in a circular economy logic.

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

[1] In 2014, 876 TWhe were produced by nuclear. This, combined with 406 TWhe 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 TWhe (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 TWhe (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

electricity produced for a given thermal power, but also less waste heat, less environmental impact and less 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 around 33%, while modern coal-fired plants reach approximately 39% and combined-cycle gas plants are even 50 to 60% efficient. 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 (or higher than) coal plants and, in the case of very high temperature, combined-cycle systems. With liquid metals as coolants this can be achieved close to atmospheric pressure, with a design that is based on an increased use of passive safety systems. This provides designers with 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 that present a long term source of radiotoxicity and heat) into short lived fission products with lower radiotoxicity. In this way, provided that high burnups are reached, the decay time of the waste can be reduced by a factor 1000 to time scales that are below 1000 years,[4] which significantly reduces the required capacities of geological repositories through an up to ten times reduced volume, much shorter-term hazard and less heat from the waste. In addition, the adoption of a closed fuel cycle requires only a short-term storage of the irradiated fuels before their reprocessing and reuse.

GenIV systems create more fuel than they burn and operate at high temperature using passive safety systems, with the possibility of burning waste: this increases substantially the sustainable use of resources and guarantees safe energy production for centuries, significantly reducing waste production.

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

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, in several cases the compatibility of materials with non-aqueous coolants needs to be demonstrated. The operating conditions for the fuel pins are further complicated by possible large temperature gradients and the internal presence of fission products. All these factors lead to substantial changes of materials' properties, which reduce their performance. At the same time, the overall cost of these systems must be on a par with other low-carbon energy systems, including current LWRs. Since the capital cost of the construction represents the largest part 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 to which both structural and fuel materials are going to be exposed. Thus, **the performance**

[4] J. Magill, V. Berthou, D. Haas, J. Galy, R. Schenkel, H.-W. Wiese, G. Heusener, J. Tommasi, G. Youinou, "Impact limits opf partitioning and transmutation scenarios on the radiotoxicity of actinides in radioactive waste", Nuclear Energy 42(2003)263

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, such as concentratred solar power (CSP), geothermal energy, bioenergy and fuel cells and hydrogen.

1.2. The EERA JPNM in the European platform landscape

The EERA JPNM provides the R&D for materials that is needed for the development and implementation of GenIV systems in Europe

The Joint Programme on Nuclear Materials, JPNM (www.eera-jpnm.eu), was launched in 2010 and is one of the 16 current joint programmes (JPs) of the European Energy Research Alliance, EERA (www.eera-set.eu). Altogether, these JPs cover all the 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 almost 250 (in 2018) 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 (TRL[5]<5). EERA deals mainly with research towards innovation, while industrial implementation (TRL>5) is characteristic of the technology platforms and the industrial initiatives.

The European Sustainable Nuclear Energy Industrial Initiative (ESNII[6]), 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 (SNETP[7]) 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 beyond this. By operating mainly at TRL<5, the EERA JPNM deals mainly with fundamental research, albeit projected towards specific technological applications, 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 2015[8] 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"

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

[6] http://www.snetp.eu/esnii/

[7] http://www.snetp.eu/

[8] http://www.snetp.eu/wp-content/uploads/2016/01/SNETP-DEPLOYMENT-STRATEGY-2015-WEB.pdf

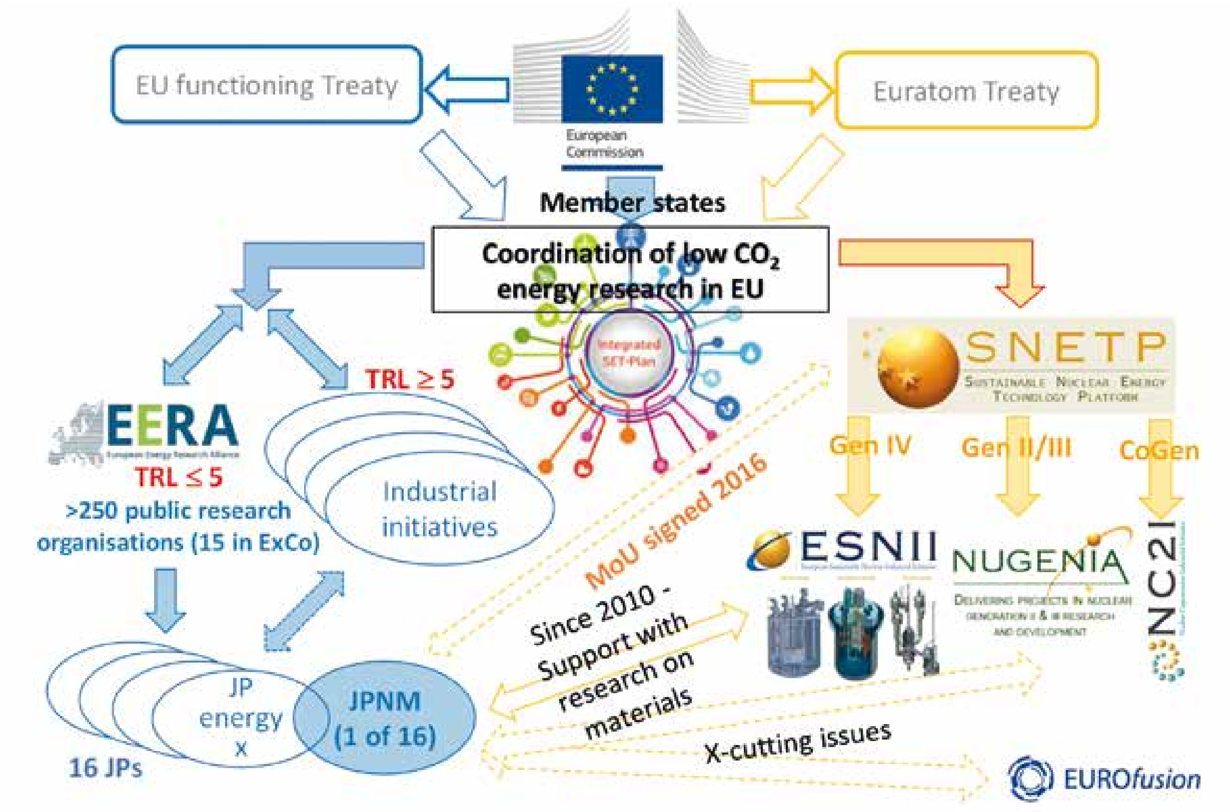


FIGURE 1: *Illustration of the connection between (nuclear) energy platforms in Europe.*

Thus, **research on structural and fuel materials' behaviour belongs to one of the areas where commonalities across nuclear reactor generations and types can be actually found**. For this reason the SNETP explicitly mentions[9] 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 cements 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)[9] and the Nuclear Cogeneration Industrial Initiative (NC2I)[10]. Moreover, several issues faced by materials for fission reactors are common to fusion reactors and systems, as well: hence it is possible to find cross-cutting topics with this other, longer-term form of nuclear energy.

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**

Research on structural and fuel materials' behaviour belongs to one of the areas where commonalities through nuclear reactor generations and types, as well as with other energy technologies, can be actually found

[9] http://www.nugenia.org/

[10] http://www.snetp.eu/nc2i/

collaboration with other joint programmes, targeting high temperature operation in environmentally harsh environments, for example AMPEA (Advanced Materials and Processes for Energy Application), CSP, geothermal energy, bioenergy and fuel cells and hydrogen.

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 deployment of GenIV fission reactors and systems, guaranteeing more sustainable and safe nuclear energy, with the potential for nuclear heat generation.[11]

GenIV reactors are expected to be commercially deployed around, or after, the middle of this century, depending on industrial commitment and political support, in particular to close the fuel cycle with fast reactors or to alternative uses of nuclear energy.[12] At the moment, four **GenIV fast reactor systems**, at different maturity levels, are being studied in Europe **within ESNII**, namely: sodium-cooled fast reactor (SFR), lead-cooled fast reactor (LFR), gas-cooled fast reactor (GFR), and accelerator-driven system (ADS).

The **SFR is the most mature technology** and **LFR is considered the next technology in terms of TRL,** with advantages in terms of passive safety features, as well as potential

Gen IV fast reactor systems considered by industrial initiatives in Europe include sodium, lead and gas cooled, as well as accelerator-driven system, technologies.

High temperature thermal reactors are also the focus of industrial initiatives.

Other GenIV systems not linked to industrial initiatives are the supercritical water and molten-salt reactors.

modularity, while **GFR is a longer term alternative** that would open the way to even higher temperatures and therefore efficiency. In ESNII, ASTRID, ALFRED and ALLEGRO are demonstrators for SFR, LFR and GFR technologies, respectively.[13] In addition, MYRRHA is a flexible research facility for material testing and demonstration of ADS for waste minimization, with features strongly related to LFR technology. **These systems are all in the focus of the EERA JPNM research**.

NC2I addresses the design of **high temperature reactors for the supply of heat to industry** (cogeneration), in order to demonstrate the feasibility and advantage of the coupling between

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

[12] For an analysis of the prospects fof GenIV systems, including technical descriptions and economics considerations, see Locatelli, Mancini and Todeschini, Energy Policy 61 (2013) 1503.

[13] At the time of preparation of the current document the ASTRID project is experiencing profound revision and may be cancelled as known now, however it keeps being used as reference SFR demonstrator throughout this document.

a nuclear reactor and a non-nuclear plant, in a framework of CO_2 emission reduction in heat production.

Despite being pushed forward by industrial initiatives, the time to construction of these European demonstrators is currently difficult to assess. It will depend substantially on industrial strategy evolution and governmental financial support, including in some cases the possibility of accessing European structural funds.

GenIV includes other reactor concepts, with specific advantages, namely: the **supercritical water reactor** (SCWR) as an advanced upgrade of existing LWRs, and so potentially easier to demonstrate in terms of technology, and the **molten salt reactor** (MSR) as a potentially proliferation-resistant system (in the thorium cycle) with advantages also in terms of fuel recycling and safety. These systems are not included in the ESNII portfolio, nor are there established industrial initiatives pushing them forward;[14] yet work on them supported by national programmes is underway in Europe . The **SCWR is included in the NUGENIA portfolio as advanced LWR**, and the MSR is increasingly mentioned as a long term interesting option. **SCWR and MSR, like the HTR, are not in the focus of the JPNM, but may be addressed in terms of materials, especially as part of cross-cutting activities**, if there is a sufficient critical mass of interest within the JPNM partners.

Importantly, **GenIV reactors should not be decoupled from the relevant fuel cycle facilities**. These enable 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 reprocessing and recycling, and together are needed to guarantee sustainability for centuries to come. For these reasons, one should generally talk of **GenIV *systems***, and not simply GenIV (fast) reactors.

1.4. Objectives and Grand Challenges of the EERA 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 service conditions, seeking predictive capability, to select the most suited materials and define for them safe design rules, especially considering radiation and temperature effects, and taking care of their compatibility with coolants.
2. **Development of innovative material solutions** to improve resistance to high temperature, irradiation and aggressive environments, through suitable processing/protective methods applied to existing materials or developing new types of materials

The EERA JPNM **vision paper**[15] expresses a view on nuclear energy materials strongly linked with the ESNII systems and identifies three Grand Challenges. These are based on the three pillars of the EERA JPNM's research approach and strategy (see Figure 2):

- Materials characterization: Assessment of candidate structural and fuel materials' behaviour in operational conditions: screening, selection and qualification with a view to developing design rules;
- Materials modeling: Development of advanced models to rationalise materials' behaviour, underpin design correlations and provide a basis for the improvement of materials properties by providing predictive capability;
- Innovative materials: Development of innovative structural and fuel material solu-

[14] This is consistent with the analysis done in the reference cited in note 13, where the feasibility of the different systems based on the state-of-the art is classified as medium or low for existing SFR and LFR concepts, but uncertain or critical for SCWR and MSR, as well as for GFR.

[15] https://www.eera-set.eu/wp-content/uploads/Vision-Paper-EERA-JP-Nuclear-Materials-February-2015.pdf

tions and advanced fabrication processes for industrial use, with superior capabilities in terms of resistance to irradiation, high-temperatures and aggressive environment.

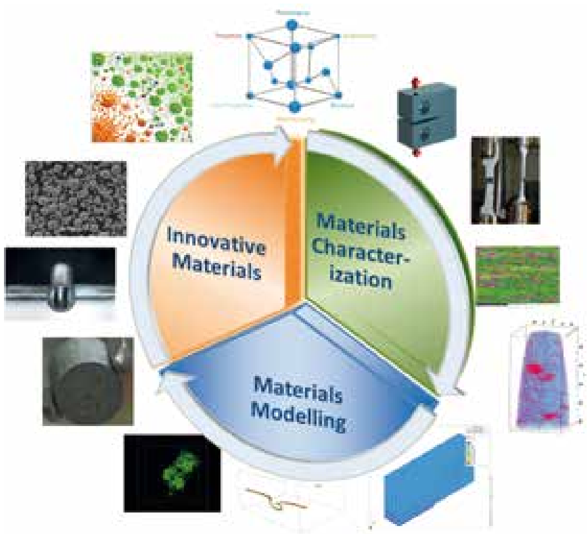


FIGURE 2: *The research approach and strategy pillars of the EERA JPNM.*

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

- **Grand Challenge 1:** Elaboration of design correlations, assessment and test procedures for the structural and fuel materials that have been selected for the demonstrators, under the service conditions expected. This involves deployment of infrastructures for exposure to ageing environments, 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. These are essential assets, given the scarcity of experimental data and the difficulty and cost of obtaining them.
- **Grand Challenge 3:** Development of innovative structural and fuel material solutions and fabrication processes, in partnership with industry, to achieve superior thermo-mechanical properties, better compatibility with coolants and improved radiation-resistance, so as to increase safety and improve efficiency and economy.

The objective of the EERA JPNM is to improve safety and sustainability of nuclear energy by focusing on materials aspects

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

1.5. *Purpose, target and structure of this Strategic Research Agenda*

The SNETP Deployment Strategy (DS) of 2015[8] provides a possible timeline for the different nuclear technologies. While the actual timeline will evolve, 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** without a deadline: it constitutes the research 'humus' on which innovation, and consequently better safety and efficiency for nuclear systems, can grow;
- In order to allow the **licensing and construction** of GenIV demonstrators, taking into account also recent developments, the research on materials needs to provide sufficient data for qualification and possibly codification of design rules, as well as computer simulation, depending on system and issue, in a horizon that has a span of **10 to 20 years**.[16]

[16] It is not possible to establish a priori when the data about a given material can be considered sufficient, because it depends enormously on the system, the component and the targeted operating conditions. This is why a continuous interaction between materials scientists and designers is needed.

This horizon, taking into account the specificities of nuclear energy, is relatively short. To be met, if the goal of sustainability is to be reached, the deployment is required now of significant resources that are devoted to materials research and development.

In this framework, the present SRA identifies **scientific and technical gaps** that need to be addressed, as well as **innovative routes** that may be followed, to provide **sufficient data, knowledge, simulation tools and experimental instruments**, concerning structural and fuel materials, to enable the **design, licensing and construction of GenIV system demonstrators,** as well as to open the way to their **longer term commercial deployment.**

This SRA identifies scientific & technical gaps and innovative routes to provide sufficient data, knowledge, simulation tools and experimental instruments, concerning structural and fuel materials, to enable the design, licensing and construction of GenIV system demonstrators, opening the way to longer term commercial deployment.

It is expected to be of use for scientists, industries and research managers and funders, decision-makers.

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

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

- The **vision paper** provides the general high-level context and is meant for any audience;
- The **strategic research agenda (SRA)**, which is revised periodically,[17] describes for stake-holders the R&D route to be followed (roadmap) to face the three Grand Challenges of the EERA JPNM identified in the vision paper, extending also to the other strategic activities that accompany research.
- The **description of work (DoW)**, regularly revised, is the implementation of the SRA: it describes work that is underway, including tasks and deliverables at the level of research subprogrammes.

The present SRA is thus expected to be of use to guide the activities of all nuclear materials stakeholders: scientists, industries,

[17] This SRA may need to be updated if: (a) materials considered have become obsolete; (b) new materials solutions appear; (c) the reference system conditions change substantially.

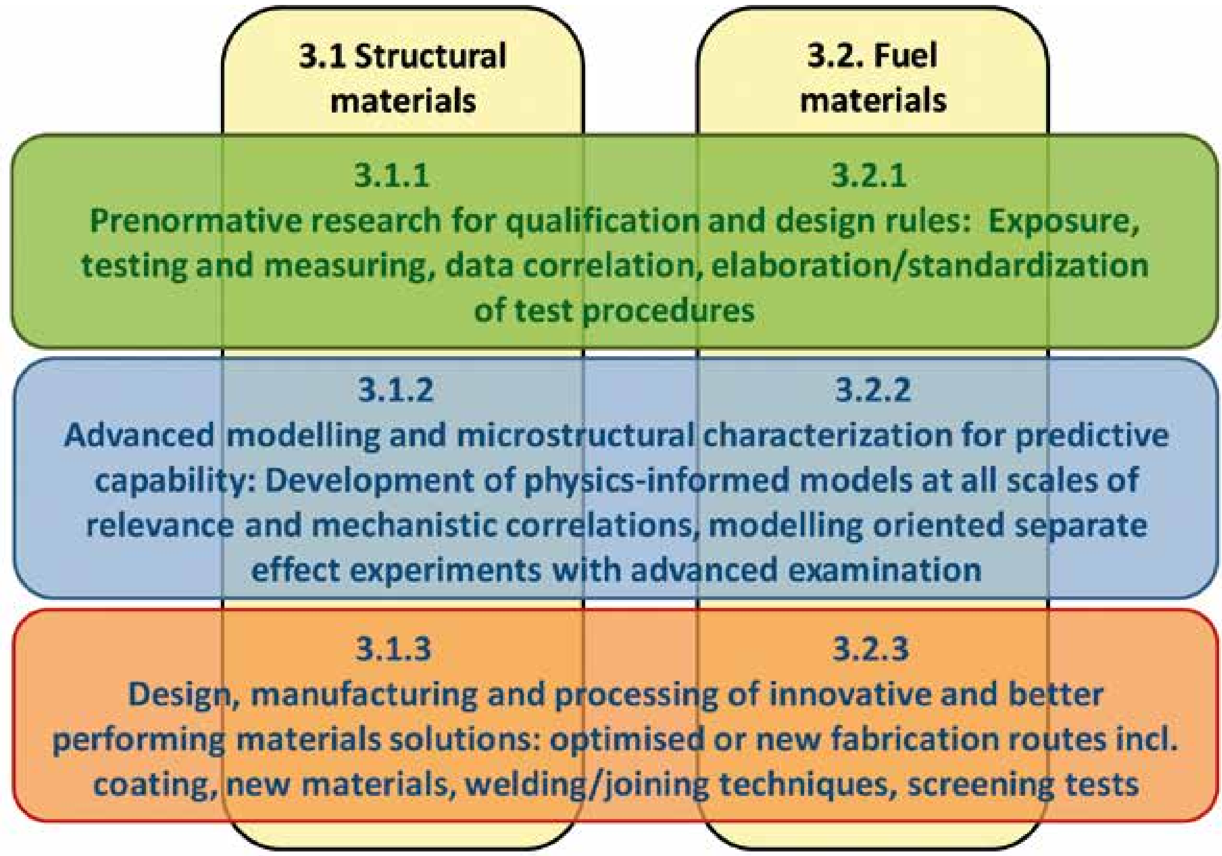


FIGURE 4: *Matrix structure of the SRA: the crossing of infratechnologies and materials identifies blocks which form the sections of the part 3 of this document.*

research managers and decision-makers.[18] It is aimed at researchers active within and without the EERA JPNM, but it has the ambition to reach other stakeholders also, 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. **Its goal will be reached if and when the industrial counterpart adopts the results of the planned work for specific component/system design**.

This SRA adopts a matrix structure (C. Featherstone and E. O'Sullivan[19]), as 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, structural and fuel, provide the vertical 'weft'[20]. Within each matrix element, or block, different issues and relevant goals are identified.

<u>The three infratechnologies that are identified are obviously interdependent</u>, as exemplified in the virtuous circle of Figure 2, which illustrates the JPNM research strategy. Prenormative research on existing materials and screening of new material solutions clearly overlap and feed each other in terms of methodology and the necessary infrastructures,

[18] http://ec.europa.eu/research/industrial_technologies/pdf/research-road-mapping-in-materials_en.pdf

[19] Featherstone and 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.

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

while physical models and simulation tools describing the behaviour of materials support and have the potential to accelerate both pre-normative research and development of new material solutions.

This document presents the research agenda, including discussions of all related strategical aspects:

- 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 and viable access schemes;
- Interaction with industry;
- Benefits of international cooperation;
- Some aspects related with education and training;
- Risk assessment.

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. Towards 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 within the six sub-programmes of the JPNM. Subsequently, contacts were established with the SNETP pillars[21] and the fusion community, as well as with other joint programmes in EERA[22]. A MoU was signed between EERA and SNETP in December 2016, with the purpose of agreeing officially on common research topics. As a 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 have been taken into account. These sources are listed in what follows.

1.6.1. Materials roadmap enabling low carbon energy technology (MR)

This roadmap was issued after a wide consultation among stake-holders that was launched by the EC in December 2011.[23] It highlighted the steps to be followed in the field of materials for advanced low-carbon energy technologies, defining a 10-year European Research and Innovation (R&I) agenda. For nuclear fission it proposed a Research and Development (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. While not all of these targets are still relevant, **several technical issues mentioned in the MR are also in the focus of this SRA, which updates them to the current perception of needs.**

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), which was officially presented and launched on the occasion of the SET-plan conference in

[21] 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.

[22] https://www.eera-set.eu/wp-content/uploads/EERA-JP-workshop-Materials_for_Energy_report.pdf

[23] https://setis.ec.europa.eu/system/files/Materials_Roadmap_EN.pdf

Nuclear energy is recognized by the SET-plan as a low-carbon energy source. Materials have clearly an important role to play to reach the SET-plan target of nuclear safety and its related Implementation Plan.

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.

Among the 10 SET-plan key actions from the IR, used as a base for discussions with Member States and stakeholders on the prioritisation of energy research activities in Europe in 2017-18, nuclear energy enters as the 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 in reaching this 10th SET-plan target and its related Implementation Plan**. The present SRA provides essentially an expansion of the IR, to allow its practical implementation for what concerns research on GenIV nuclear materials.

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

The OECD/NEA Technology Roadmap for Nuclear Energy 2015[24] 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. **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. SNETP Strategic Research and Innovation Agenda and Deployment Strategy

The SNETP released a Strategic Research and Innovation Agenda (SRIA)[25] 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;

[24] https://www.oecd-nea.org/pub/techroadmap/

[25] http://www.snetp.eu/wp-content/uploads/2014/05/sria2013_web.pdf

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, cogeneration, and also fusion.

Consistently, in the present SRA, significant effort is made to identify cross-cutting issues with other energy technologies.

- 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 a key message 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. (Similar effort is made for fusion energy and other energy technologies). 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 used to identify overlaps and set boundaries.

1.6.5. GIF Technology Roadmap[3] and R&D Outlook

The Generation IV International Forum (GIF) was founded in 2001 with the objective to establish international co-operation in R&D for six future nuclear systems (SFR, LFR, GFR, SCWR, VHTR, MSR).

The Technology Roadmap was published in 2002 and updated in 2013; a R&D outlook was published in 2009[26]. The Roadmap and R&D Outlook provide an overview of the major R&D objectives and milestones for the coming decade, aiming to achieve the Generation IV goals of sustainability, safety and reliability, economic competitiveness, proliferation resistance and physical protection. The emphasis is on the systems for which materials is an integrated part.

1.6.6. Other roadmaps

Other documents connected with the present SRA in terms of issues, materials and goals, that were consulted during its preparation, are: (i) the SRA of EuMaT, the European Technology Platform for Advanced Engineering Materials and Technologies[27]; (ii) the roadmap of Materials Modelling of the European Materials Modelling Council[28]; the Roadmap of the Metallurgy Europe – EUREKA cluster[29]; the Roadmap to Fusion Electricity (2012 version)[30].

[26] https://www.gen-4.org/gif/jcms/c_42188/publications

[27] http://eumat.eu/filehandler.ashx?file=11580

[28] https://emmc.info/wp-content/uploads/2015/04/EMMC_Roadmap_V3.0.2.pdf

[29] http://metallurgy-europe.eu/

[30] https://www.euro-fusion.org/fileadmin/user_upload/EUROfusion/Documents/Roadmap.pdf

2

NUCLEAR MATERIALS FOR GEN IV SYSTEMS AND RESEARCH APPROACHES

2. NUCLEAR MATERIALS FOR GEN IV SYSTEMS AND RESEARCH APPROACHES

2.1. Material degradation processes and relevant properties

The 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.

Liquid-metal cooled system **operation temperatures** are expected to be between 400-550°C for demonstrators, with off-normal **excursions** up to 600°C (unless new materials allow to go higher), but ideally targeting 600-700°C or even beyond for commercial plants, while operation temperatures in gas-cooled systems target 850°C or beyond. In the latter, this will lead to temperatures around 2200°C at the centre of the fuel in normal conditions, and temperatures may exceed 1000°C in structural materials in off-normal conditions. These temperatures, coupled to tremendous **temperature gradients** (up to 500°C/mm), will inflict severe **thermal and mechanical stresses** on the fuel and plant components that will act to cause 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. These effects are unavoidably exacerbated by high temperature. Thus, what often **limits the operating temperature** is not only the degradation of the properties of the materials *per se*, but the **effect on the materials of the aggression by heat transfer fluids above a certain temperature**. Finally, nuclear materials are exposed to varying levels of **irradiation**: for example, the dose measured in number of displacements per atom (dpa) will reach 1 dpa in the fuel in less than 1 day in reactor, will exceed 100 dpa in the cladding during its stay in the reactor, and may be less than 2 dpa in the vessel over the whole reactor life. Invariably, these levels of **irradiation will negatively affect the performance of the materials**.

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

For **fuel**, one of the main parameters that needs to be maintained under irradiation is the temperature at the centre of the pellet. This is governed by the thermal conductivity and needs to be less than the fuel **melting temperature**, itself strongly affected by the irradiation-induced modifications of the fuel within the reactor (creation of defects and fission products, restructuring). Secondly, the integrity of the fuel pin, which is affected by the increasing mechanical load on the cladding as burn-up increases and by the possible chemical interaction between fission products and cladding elements, must be maintained in order to prevent fission product release.

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

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

2.2. Nuclear materials, components and systems

2.2.1. Structural materials

Structural materials are those used to fabricate a component that bears load or stress, whatever 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 (relatively) easily extracted from the reactor, or their replacement is a normal part of the reactor operation. The most obvious examples of this type are the fuel elements, which are periodically reshuffled or removed once the burnup limit allowed by neutronics and materials has been reached. In general, these elements are the hollow tubes that contain the fuel pellets, the bundles that hold the tubes together and the structures that support these. A significant deviation from this general description of the fuel element may be found in HTR or GFR, where the high temperatures may require the use of ceramic materials.

  Replaceability generally goes hand in hand with the expectation of more severe degradation and therefore shorter lifetime. This is the case for the **fuel cladding**, which is exposed to the highest irradiation dose, experiences the highest temperatures and also high temperature gradients, being in contact with the coolant and the fuel.

- Non-replaceable components constitute the main structure of the reactor. Major examples of this type 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.[31] These components need to be designed for the full lifetime of the reactor or, correspondingly, 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 sufficiently slow to guarantee that the component remains fit for purpose until the end of the life of the plant. Typically, the vessel will be sufficiently distant from the fuel elements to be subject only to marginal or negligible irradiation and the system will be designed so that the vessel temperatures will be lower than close to the fuel element. In addition, it is sometimes possible to apply protection the vessel 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 obviously comes, to a large extent, from previous experience. **The current design of GenIV demonstrators planned in Europe envisages the use of austenitic steels as the main class of in-core structural materials**, specifically 316L(N) for most components, including the vessel, and 15-15 Ti for the cladding and other fuel element parts[32]. Depending on the specific demonstrator, other materials may also enter, e.g. **ferritic/martensitic (F/M) steels** in the SFR core, while for the GFR core **ceramic materials** similar to those used in HTR, or perhaps **refractory metallic alloys**, must be considered. Other materials foreseen for use in **out-of-core** components are **Ni-based alloys**, especially for the intermediate heat exchanger, the steam generator, turbine blades, coaxial pipes or hot gas headers of GFR and HTR. However, **austenitic steels are clearly the current dominant choice across system demonstrators**. The reason is that they are a very good compromise between several requirements, even without excelling specifically in any one of them. The critical reason for this choice is the **return of experience** from their use in the fast reactors built and operated in the past, such as Phénix and Superphénix in France. There exists, therefore, a wealth of experimental data on them, on the basis of 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**. Therefore, the **largest effort of qualification foreseen in this SRA is dedicated to austenitic steels** (section 3.1.1). Research activities also include **paths to *improve* the properties of austenitic steels** (section 3.1.3): (1) improve swelling resistance; (2) improve corrosion resistance, especially against HLM. Higher **swelling resistance** is pursued via microstructure stabilisation of the steels with suitable alloying elements (e.g. AIM1 and its improved version AIM2). Better corrosion resistance is pursued with the creation of **corrosion barriers**, either by changing the surface composition with the implantation of aluminium to form a protective alumina layer, or by directly applying alumina (or other ceramics) coatings on the metallic substrate. Another possibility is to add aluminium directly to the alloy, to allow the formation of a self-healing protective alumina layer when in contact with oxygen-containing fluids: these are the so-called **alumina forming austenitic** (AFA) steels, which would also exhibit inherently good creep properties due to NiAl precipitation. None of these solutions, however, is currently included in design codes.

F/M steels (e.g. T91, EM10, HT9, reduced activation F/M like EUROFER or F82H for fusion,…) **potentially offer a number of desirable properties as cladding and core materials** that are

[31] 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.

[32] These are AISI and customarily used names. In Europe the formally correct way to denote them is X2CrNiMo 17 12 2 controlled nitrogen content (for 316L(N)) and X10CrNiMoTiB 15 15 (for 15-15 Ti); the latter, when of German production, is also denoted as 1.4970.

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 of irradiation dose,[33] F/M steels are **expected to reach 200 dpa without experiencing any swelling**, thereby offering a promising solution to significantly increase the burnup. However, F/M steels can currently operate only below 550°C, suffer from radiation-induced embrittlement below 350-400°C and are especially prone to liquid metal embrittlement (LME), when in intimate contact (wetting) with, in particular, heavy liquid metals (HLM)[34]. Reliable **correlations and models for the identification of design rules** allowing for all these phenomena are **needed to make F/M steels** usable in GenIV systems; alternatively, the resistance to high temperature and embrittlement of F/M steels, as well as the compatibility with HLM, need to be improved.

Two paths are pursued to **improve the creep 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 thermal-mechanical treatments (TMT) in a conventional metallurgy framework (**creep-strength enhanced**, CSE). The use of ODS F/M cladding would allow the upper limit of the temperature window to be increased beyond 700°C, improving radiation resistance as well. However, the ODS steel fabrication process is costly and there is a lack of established industrial production. The second way to improve the properties of F/M steels, via composition tuning and thermal-mechanical treatments (TMT) to stabilise the microstructure, is very attractive because the manufacturing process remains within conventional metallurgy and has been used successfully outside the nuclear field (650°C operation limit). Similar techniques may also help to limit radiation embrittlement below 350°C. The methods to **improve corrosion resistance** are the same as in austenitic steels: either surface modifications/coatings or alloy composition tuning with aluminium addition: **FeCrAl alloys** may also be ODS, giving rise to a potentially very promising material that is resistant to irradiation, creep and corrosion.

For **temperatures above 800°C**, such as those **targeted in GFR and HTR**, no steel is able to maintain its fitness for purpose, with the possible 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 quite severe irradiation embrittlement and also swelling, so their use in the core can be critical. Other **heat-resistant materials** need therefore to be considered for in-core applications and the spectrum is quite wide, ranging **from refractory metals (e.g. molybdenum or vanadium) to ceramics**. The latter generally offer 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 composite is SiC_f/SiC**, i.e. silicon carbide fibres in silicon carbide matrix, and SiC_f/SiC is the main candidate material for GFR cladding. It is also a material of use for the HTR and it is a suitable candidate for LFR and SFR cladding, as well. However, this material is very expensive and design rules for it are not developed.

Moreover, the design of the high temperature gas cooled reactor concept (GFR, V/HTR), imposes the need for materials solutions to be devised for the purposes of **thermal shielding and insulation**, as well as **control rods and seals**. These materials must be gas-tight, corrosion resistant, and exhibit high fracture and creep strength, while being inexpensive and

[33] Currently, 15-15 Ti is supposed to reach 90 dpa, better versions of it (AIM2) might allow this limit to increase up to 110 dpa.

[34] Liquid lead and its alloys, such as Lead-Bismuth Eutectic (LBE) or Pb-Li, are collectively denoted as heavy liquid metals (HLM). Recently it has been observed that also austenitic steels in liquid sodium may experience LME, although the conditions for this to happen are not easy to comply with.

TABLE 1: *Main structural materials expected to be used in the different ESNII GenIV systems, distinguishing different phases that define the timeline, from demonstrator to prototype (FOAK), and finally commercially deployed plants.*

PHASES → / SYSTEMS ↓		ESNII demonstrator		FOAK (prototype)	Commercial deployment
		As licensed (phase I)	**Evolving (phase II)**		
SFR (ASTRID)	Periodically Replaced Components	AuSS: 15-15Ti – AIM1 (cladding) F/M: EM10 (wrapper)	AIM2 or F/M ODS (cladding)	TMT F/M or 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	
ADS (MYRRHA)	Periodically Replaced Components	Cladding: 1.4790; structures: 316L(N)	Coated 15-15Ti (FeAl, FeCrSi, FeTa, MAX phases, ...) or AFA	N/A	
	Permanent Structural Components	316L(N)			
LFR (ALFRED)	Periodically Replaced Components	Cladding and structures: (Al_2O_3 coated) 15-15Ti (AIM1)	Cladding and structures: Al_2O_3 Coated 15-15Ti or AFA	Cladding: AFA or FeCrAl ODS Structures: AFA	AFA or FeCrAl ODS, or (coated) Mo-ODS, or SiC_f/ SIC ,
	Permanent Structural Components	316L(N)		AFA or ferritic steel lined with AFA	
GFR (ALLEGRO) / (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	GFR : 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	

preferably fabricated in a net-shape design, with the ability to be joined. At the moment, in addition to SiC, also other materials such as carbon composites, mullite, Al_2O_3, TiO_2, ZrC, ZrN, B4C, graphite or graphene are being considered for this function.

Perspective ceramic or metallic refractory materials, including innovative ones, that can be considered for cladding or other applications in GFR/HTR, and also for other systems like LFR to mitigate corrosion effects, are: ODS-Mo, high entropy alloys (HEA), and MAX phases.[35] 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.

35 Most likely suitable as a corrosion protective coating at not-too-high temperatures.

Based on this excursus on structural materials for GenIV systems, Table 1 attempts a classification and a prediction of the main structural materials that are expected to be used in the different GenIV systems that are currently part of ESNII, distinguishing different time phases from demonstrator (phase I and II) to prototype first-of-a-kind (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.** The actual timing for each system is not the same and will crucially depend on the priorities that are 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 schematized in the table remain valid and therefore, **through this table, it is possible to establish priorities and perspectives**. Table 1 is not expected to be fully comprehensive, but simply highlights which classes of materials the EERA JPNM research focuses on and where they fit, in order to immediately understand the application and context. This table is based on current knowledge and perceptions about the future and will require updates. Phases I/II of the prototypes are not necessarily defined in the ESNII plans and can be of very different nature, but essentially correspond to switching to different materials, particularly cladding and fuel element materials, for better performance. For instance, the GFR considers a first demonstrator working at low temperature (~500°C) to be later upgraded for high temperature operation (~850°C), when the qualification of the necessary materials is sufficiently advanced; a similar strategy is envisaged for the LFR demonstrator (380→520°C). Finally, concerning FOAK and commercially deployed reactors, the materials mentioned in the table are obviously an educated guess. The message is that materials that are not currently envisaged for the demonstrators remain a target of the EERA JPNM research, because of their long term potential.

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 burnup 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, while ensuring dimensional stability within the design margins, 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. Different geometries of 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 neutron reactor prototypes, although UO_2 with <20% U-235 is planned to be used in the first core of ALLEGRO. Variations of MOX were used in the previous European fast reactor programmes: Phénix, Superphénix, 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 significant 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 and later for Gen IV reactors. First, the MOX product can bear important intrinsic hallmarks that are linked to the fabrication methods used, e.g. porosity distribution, grain size, and impurity levels, all of which have a big impact on its performance. Furthermore, reactor core designs have evolved, with particular effort based on thermal hydraulics and neutron physics to reduce safety relevant parameters as void coefficients, so pellet geometry has evolved from the classical one to thicker pellets with an annulus to limit centre line temperatures. Furthermore, the GEN IV 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 accelerator-driven systems also present specific issues that may affect the fuel pellets, for example due to the multiple beam trips, which will affect power and might create temperature oscillations, so fatigue effects could be substantially more serious than in a critical reactor.

The introduction of multi-recycling of Pu will also perturb the Pu concentration in the fuel, but more importantly higher concentrations of ^{241}Pu in the Pu isotopic vector will lead to higher contents of ^{241}Am, 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 also on the in pile performance as helium will be produced in the fuel 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**

TABLE 2: *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.*

PHASES → / SYSTEMS ↓	ESNII demonstrator		FOAK (prototype)	Commercial deployment
	As licensed (phase I)	Evolving (phase II)		
SFR (ASTRID)	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	
ADS (MYRRHA)	MOX with up to 30-35% Pu	MOX, MA bearing fuels	N/A	
LFR (ALFRED)	MOX with up to 30% Pu	MOX, advanced MX	Advanced MX, MA bearing fuels	
GFR (ALLEGRO) / (V) HTR	UO2 with <20% U-235	MOX, advanced MX	Advanced MX, MA bearing fuels	

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, though, if high purities are to be achieved. There are questions about their volatility at temperatures below the melting point; this is a matter that increases in significance for Pu multi recycling, as the built in Am component could be (but it is not fully proven) even more volatile than the U and Pu constituents.

Similarly to Table 1 for structural materials, **Table 2** 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 1.

2.3. Materials research approaches

One of the main duties of materials engineers is to be able to foresee the time when, due to degradation, the material is no longer fit for its purpose under any kind of external influence, thereby defining the lifetime of systems and components. This implies **knowing what happens to the material and how its properties change while in operation, as well as in off-normal conditions** that may lead to accidents, 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. This knowledge is also essential to identify the limits of materials, often leading to a redefinition of the system and component design, to ensure that the operation conditions are such that the materials remain fit for purpose for a sufficiently long time.

The acquired knowledge of the behaviour of a material is then transferred to **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 and monitoring cannot be performed**, or to reduce the need for testing and monitoring, knowing that in the event the materials in operation can, in some cases and within limits, be monitored using non-destructive techniques, but cannot be continuously tested. Models underlie the design rules and codes and provide a guide to plan the maintenance, inspection and, if needed, replacement of components, based on the assessment of their lifetime or probability of failure. Modelling is therefore of crucial importance, although the modern trend is to **enhance the materials monitoring capabilities**, in particular through **digitalisation** (big data analysis, use of artificial intelligence, …), in order to reduce the need for testing and to complement the model indications.

As an alternative to changing design conditions, which in some cases may penalise the efficiency or the economy of the system, the knowledge of the behaviour of materials leads naturally to the identification of ways to improve their response under given conditions. This can be done by protecting them, or improving their properties through appropriate processing, or by identifying or even developing in a targeted way entirely new materials. The **development of new material solutions allows design constraints to be relaxed, thereby improving, sometimes substantially, the efficiency or the economy of the system**.

These three aspects of materials research, namely qualification, modelling and new materials solutions, 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. They are also linked and constitute a virtuous circle, as illustrated in Figure 2. They are described 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

GenIV system demonstrators will necessarily be constructed using structural and fuel materials on which some return of experience exists from previously operated reactors, and which are possibly already included in design codes. However, innovative design such as for the **ESNII systems can generally rely only on limited return-of experience.**[36] Thus, safety and other requirements need to be demonstrated by **qualifying the materials selected for the operating conditions expected in the demonstrators**,[37] to which they need to be proven to resist for sufficient time. These include high temperature, high irradiation levels, and compatibility with coolants, for instance between HLM and structural components.[38] In turn, the actual **operating conditions of 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 now imposed and more effort to demonstrate structural integrity in accident scenarios is needed.[37]

[36] 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 or any advanced SFR demonstrator 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.

[37] In design codes, e.g. RCC-MRx, operating conditions can be defined as: (1) conditions to which component may be subjected in the course of normal operation, including normal operating incidents, start-up and shutdown; (2) emergency conditions corresponding to very low probability of occurrence, but which must nonetheless be considered, and which imply shut down and appropriate inspection of the component or of the plant; (3) highly improbable operating conditions, whose consequences on components are studied among others for safety reasons. Severe accident conditions are not necessarily part of design codes, although In the aftermath of Fukushima this is being discussed ("Design Extended Conditions").

[38] 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 (by oxidation and dissolution) and liquid-metal embrittlement.

Such accident scenarios generally translate into load excursions with high temperatures and load rates. The temperature range for mechanical properties assessment needs therefore to be extended and strain rate properties are necessary in the relevant range. When such data do not exist, specific research programmes are required.

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.[39]

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,[40] 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, which may be 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 that 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 requirement**. In some cases of interest for GenIV systems, the characterisation may necessitate the identification of new and bespoke standard procedures to execute the exposure and, especially, the tests. The procedures are particularly critical in the case of accelerated tests, because their relevance to real operating conditions needs to be proven.

In-pile experiments, in principle, imply repeating the same type of exposures under **irradiation,** and preferably in **fast neutron spectra** for 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, although irradiation is limited to lower dose, so that the possibility of safely extrapolating to different spectra and higher doses is essential.[41]

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 from scattered material data for fatigue curves or accelerated laboratory creep data are extrapolated to the operational conditions that correspond to lower stresses and temperatures. Design rules eventually comprise closed-form equations and strict criteria, together with basic material properties. They are

[39] 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.

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

[41] 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.

developed to be conservative and simple to apply at the design stage. An important concept is "allowable stresses"; these 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**,[42] together with materials specifications & design data for different materials and components, by making reference to test standards and qualification procedures.

The RCC-MRx design 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 also includes materials manufacturing specifications.[43]

RCC-MRx does not contain any specific rules for environmental effects except thinning (by corrosion), nor does it cover, yet, the high temperature area for GFR and (V)HTR; moreover, material property curves and design rules are based on 40 years operational life. RCC-MRx is updated every three years. Other codes include the R-codes developed to support the plant life management of the UK reactors; R5 is for high-temperature problems and R6 is primarily for defect assessment. There are some commonalities between the RCC-MRx appendices (note 44) 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 and other parts, the main design tools are **fuel performance codes** (FPC) that 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.[44] 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

[42] The two main design codes are RCC-MRx developed by AFCEN and ASME Boiler and Pressure Vessel Code 2015 Edition. The R5 for high-temperature and R6 for defect assessment developed in the UK are the most used assessment procedure codes in Europe.

[43] RCC-MRx 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 addresses design and construction. 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 for 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. This corresponds to ASME's "Code Cases".

[44] Fuel assemblies generally need full qualification in-pile and are not included in design codes. There is an AFCEN code for fuel assemblies, RCC-C, "Design and Construction rules for Fuel Assemblies of PWR Nuclear Power Plants", but it is restricted to PWR and it is essentially a synthesis of best practices of the French industry for fuel fabrication, with, however, very few quantitative criteria.

fuels and/or in specific operating conditions and to demonstrate the satisfactory behaviour of fuels in all operational situations to support safety reports.

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 them. 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[45] developed by JRC in collaboration with various institutions through Europe and the GERMINAL code[46], which is part of the PLEIADES platform[47], 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, data correlation, elaboration/&standardization of test procedures", 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 essential, but may not be sufficient or affordable. The exposure to real conditions in the laboratory is challenging in terms of infrastructure, time, costs, and also know-how and the data collected cannot cover all possible conditions. In particular, exposure times comparable with the lifetime of the reactor,[48] 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, extrapolation of data is unavoidable, but purely empirical extrapolations may be difficult or even unreliable. 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 is still used today and will continue to be used, but is **gradually undergoing a paradigm shift**, whereby the materials are subjected to **"design and control"**. This paradigm shift is expected to **mitigate costs and reduce lead times to deployment**.

At the heart of the design and control paradigm lies a **greater reliance on advanced modelling and simulation**, partly generated by improved theory, but also by the vastly increased computational power of the last decades, crucially **coupled with advanced microstructural and micromechanical characterization**, using ever more powerful techniques for materials examination and testing at all scales. This approach, lately denoted as integrated computational materials engineering (ICME),[49] bears the promise of providing increased robustness in the long term, initially only underpinning, then gradually improving and finally replacing traditional empirical approaches, such as those

[45] Lassmann, TRANSURANUS: a fuel rod analysis code ready for use, J. Nucl. Mater. 188 (1992) 295; Di Marcello et al., Extension of the TRANSURANUS plutonium redistribution model for fast reactor performance analysis, Nucl. Eng. Design, 248 (2012) 149.

[46] Lainet et al., Current status and progression of GERMINAL fuel performance code for SFR oxide fuel pins, IAEA-CN-245/222, FR17, Yekaterinburg, Russia, 2017; Lainet et al., Recent modelling improvements in fuel performance code GERMINAL for SFR oxide fuel pins, IAEA-CN-199/241, FR13, Paris, France, 2013.

[47] Marelle, Validation of PLEIADES/ALCYONE 2.0 fuel performance code, Water Reactor Fuel Performance Meeting, Jeju, Korea, 2017.

[48] 1/3 is considered sufficient for extrapolation, but this can also be a challenging duration/dose.

[49] ICME-based microstructure-based description of the deformation of metals: theory and application, in: Helm, Butz, Raabe and Gumbsch, Microstructure-Based Description of the Deformation of Metals: Theory and Application. JOM 63 (2011) 26. http://www.dierk-raabe.com/icme/

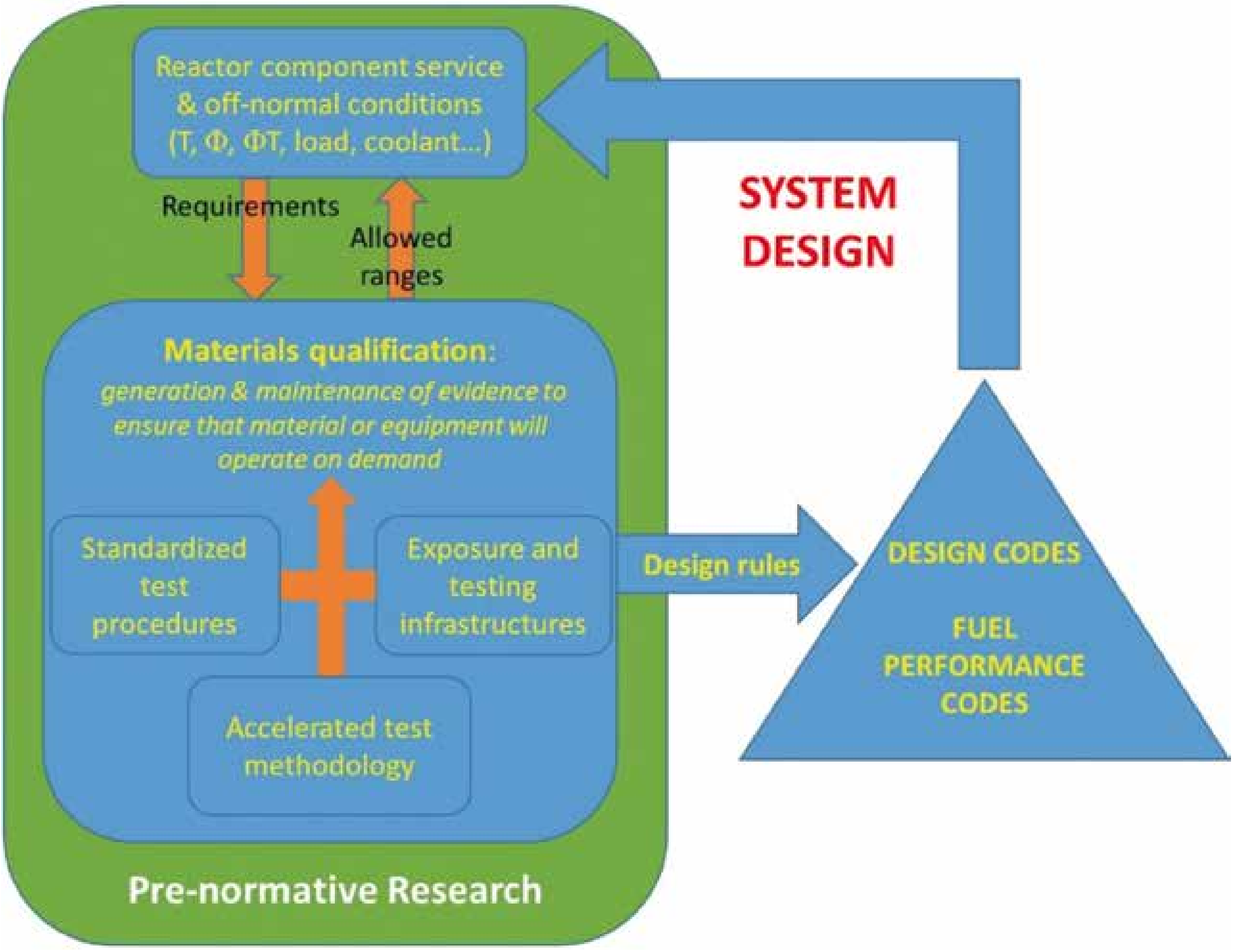


FIGURE 5: *Schematic representation of the main elements that enter the process of generation of component design codes and fuel performance codes.*

used in current fuel performance codes or in dose-damage correlations for LWR vessels.

The ICME approach aims to reach **predictive capability** by being able to describe in a **physical way**, possibly with a single tool (e.g. a Monte Carlo or a phase field code) fed by more fundamental models or calculation methods (e.g. density functional theory), 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 and/or 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, in a multiscale modelling framework and spirit.[50] For example, in order to bridge from the microscopic description to higher scales to inform continuum mechanics, models that describe dislocations (involved in all plastic deformation processes, from tensile

[50] As an example, here is an idealised chain 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, ...

properties to creep) are generally needed.[51] The modelling tools are generally costly-to-run and often use parallelized computer codes, so the use of **high performance computing** (HPC) is expected to be a crucial asset in many cases.

All these models require suitable **data** for **validation** and also **calibration** from **modelling-orientated experiments**. These are experiments in which materials are exposed to external factors, as for qualification purposes, but are designed to understand mechanisms, by **separating variables and effects**, rather than to measure engineering properties. Experiments of this type, in which key variables such as temperature, irradiation-dose and dose-rate need accurate control and variation over sufficiently wide ranges, are invariably delicate to perform, and may be long and costly. As in the case of qualification, specific facilities are needed for exposure, in particular for irradiation and subsequent characterisation. In this context, the use of **charged particle irradiation** (ions, protons, electrons, …) is a very valuable and affordable tool.

To fully characterise the microstructure of the materials exposed, the use of **combined modern characterization techniques** is crucial, e.g. transmission electron microscopy (TEM) in all of its multiple forms,[52] atom probe tomography (APT), small angle neutron scattering (SANS), positron annihilation spectroscopy (PAS), and many others. Suitable mechanical characterization experiments are equally crucial, including **micromechanical characterization from specimens at single grain scale** (nanoindentation, micro-pillars, …) with different crystal lattice orientations. Micromechanical characterization techniques are often the only possibility for charged particle irradiated specimens, due to the limited penetration of these particles. In addition, mechanical tests addressing **uni- vs multi-axial load, cyclic load, relaxation, load sequence, non-proportional loading,** etc. are of interest.

Since the requirements of experimental precision, accuracy and completeness for model validation are extremely high, it is important to establish, for all the characterization techniques used, accepted **best practices, protocols** and possibly **standards** for their application and for the analysis of the results. This should allow full inter-laboratory comparability and provide high guarantee of reliability, to reduce scatter and uncertainty. The analysis of characterization results is often helped by models and software, often irreplaceably: these models should be improved, enlarging their physical bases, and the codes benchmarked, in order to estimate and once again reduce scatter and uncertainty.

Advanced computational tools such as **artificial intelligence/machine learning** (AI/ML) can be very valuable in this modelling framework, either to help bridge through scales, whenever the complexity of a lower scale (for example chemical complexity or microstructural complexity) needs to be reduced to a limited number of variables that affect the higher scale, or to by-pass model chaining, through training on examples obtained from complex models. Additionally, these

[51] For example the following combination 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.

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

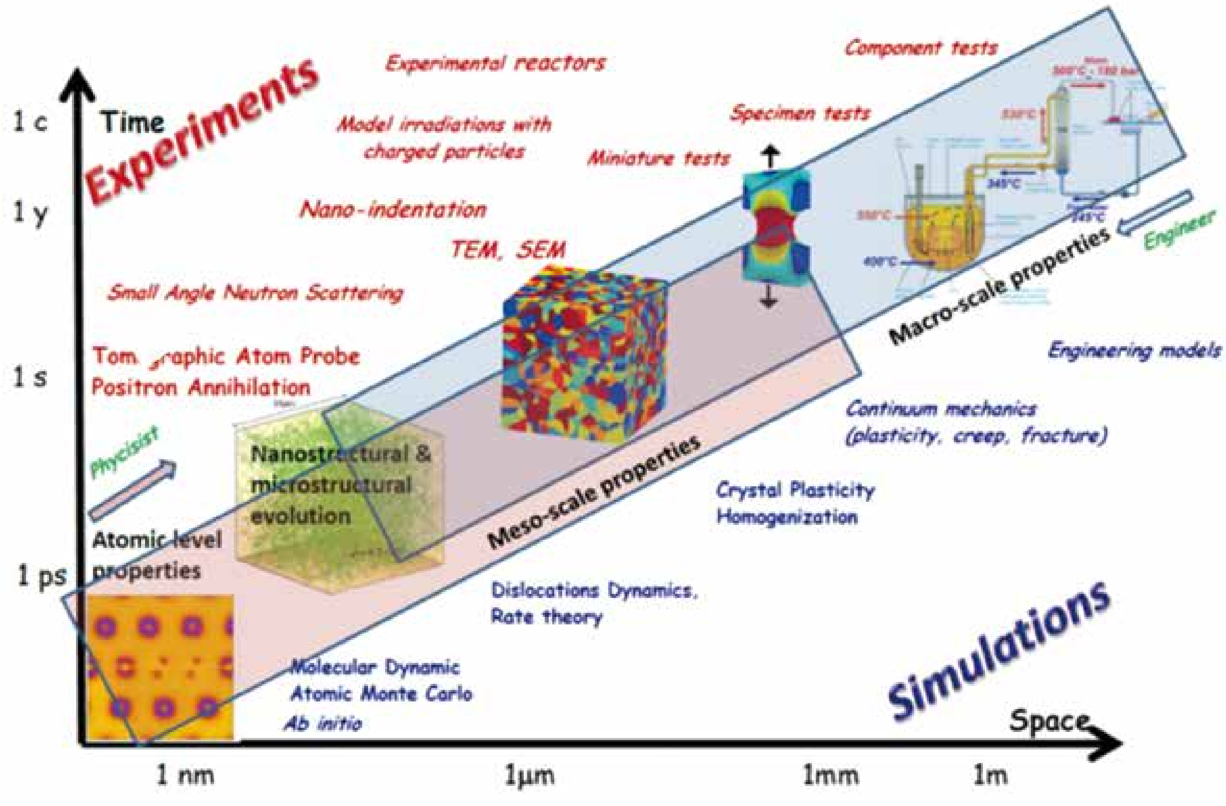


FIGURE 6: *Modelling and experimental characterisation techniques at various time and length scales involved in the ICME approach, used to elucidate the basic mechanisms of nuclear material behaviour and ageing, and their coupling.*

methods can provide support to the analysis of experimental data. In particular AI/ML is a very powerful technique for the **analysis of big data**, e.g. from extensive experimental measurements, in order to deduce a logic to be used to guide or underpin models and design correlations, but also to improve fabrication routes or materials compositions and/or architectures (see section 2.3.3). Similarly, AI/ML may assist online non-destructive monitoring of materials and components to help predict failure.

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 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**.

2.3.3. Development of advanced material solutions

For some components or parts of components, there may be no design-driven ageing mitigation strategy identified that fully guarantees the materials' fitness for purpose over a sufficiently long timespan, thereby penalising the efficiency and/or the economy of the system. The component may then have **reduced availability** (because of the requirement of frequent inspections) or **too short lifetime** (because of materials' expected rapid degradation), or offer too **little safety margin**. In this case there is a strong push to find **new material solutions** instead of, or in addition to, design solutions. A further push towards new materials solutions comes from the need to **reduce the cost** of material or component fabrication and also to **enhance the economy** of the plant within **ample safety margins** by further reducing the frequency of inspections and therefore shutdowns and increasing the lifetime of components.

New material solutions may correspond to: (i) select a **different existing material of the same class** with better properties for a specific application; (ii) suitably **protect the selected material** with functional materials to reduce its degradation; (iii) **improve the inherent properties** of existing materials or classes of materials, either by changing their composition (e.g. alloys) and/or architecture (e.g. composites), by applying different post-fabrication treatments (e.g. thermal treatments in steels), or else by using different fabrication processes; and (iv) select from an **entirely different class of materials** or, if possible, **develop new materials in a targeted way**.

This situation applies specifically to commercial GenIV reactors. It is very unlikely that the high fuel burnups and operating temperatures targeted in these systems, which are in contact with non-conventional coolants, can be fully achieved with existing materials under the overarching principle of safety. It is well-known that **GenIV demonstrators will necessarily operate at conditions that are not optimal in terms of efficiency and waste minimisation**, and at smaller scale and power than eventual commercial reactors. This is the only way to be compatible with the conditions that materials can be expected to tolerate, based on the existing return of experience and the available qualification data. Moreover, protection against degradation (e.g. against corrosion) through materials solutions is likely to be needed for the demonstrators. **To ensure the commercial deployment of truly GenIV systems with a design lifetime of 60 years and able to sustain high fuel burn-ups and transmutation, new material solutions are needed**.

A **substantial research and development and innovation effort** is required, leading to the identification of effective material solutions to mitigate the consequences of harsh service conditions, the improvement of the properties of existing materials through new fabrication or treatment processes and, if possible, the elaboration of completely new materials.

The elaboration of new materials solutions is, as always, largely based on previous experience: often via the adaptation and application to a different technology of solutions that have already been successfully used in other technologies where similar problems had to be faced. For instance, the application of coatings is a quite obvious solution to protect against environmental aggression (corrosion); and procedures successfully used to improve the creep strength at high temperature of steels used in gas- or coal-fired plants may also appear as a good solution for components that are subjected to irradiation. However, the **transfer of solutions from other technologies** must be done **within the constraints of nuclear applications**, where **irradiation** represents a very specific type of load on materials (either for property degradation or transmutation/activation) and **safety requirements** are particularly strict.

Each time a new materials solution, as opposed to a design solution, is proposed for a nuclear reactor, a **long process of full material qualification and codification** is required. Irradiation in suitable facilities and subsequent testing under all those conditions that may put safety in question, makes the process currently very long, of the **order of decades**. Moreover, before a new material or material combination is introduced, efficient **procedures for joining** pieces made of that material need to be developed and also tested and qualified. Finally, materials solutions and joining procedures are typically developed in the laboratory: crucially, before the solution is adopted there needs to be an **industrially upscaled** fabrication and joining procedure. Clearly, these steps can only be taken for a very reduced number of materials, which have emerged from a selection based on a previous **screening**.[53]

[53] A positive consequence of the strictness of the qualification procedure that needs to be undergone for nuclear application, combined with the especially harsh conditions that materials face in nuclear reactors, especially GenIV systems, is that the materials solutions developed in this framework, where safety is an imperative, are likely to be of use also for other technologies where harsh conditions are expected.

The screening is currently performed essentially in the same way as for existing materials that need to be qualified for the demonstrators, i.e. by exposure and testing (the "observe and qualify" paradigm). The only difference is that the first goal here is not to fully define the correlations and rules that will allow the design, licensing and construction of a reactor, but only to give a **first assessment of the behaviour of the candidate materials** to identify the most suitable one(s). It would be very desirable for the screening process to be done at an **accelerated pace**. Accelerating the screening of materials is very much connected with the shift towards a "design and control" paradigm. This implies addressing two aspects: (i) identifying experimental methods to **accelerate the exposure and subsequent testing** and characterisation of materials; and (ii) making use of **advanced modelling and characterization techniques to guide the development** of new materials, by using a targeted quantitative methodology rather than a trial-and-error approach that is based on the experience or intuition of the researchers involved.

In the case of nuclear materials, **charged particle irradiation** can be a suitable method to accelerate exposure, also in combination with other parameters (temperature and environmental attacks), provided that: (i) suitable characterization techniques, including for mechanical characterization, are identified, developed and, importantly, standardized or at least guided by accepted best practices and protocols; and (ii) models are developed that allow the assessment of the effect of irradiation by particles that are different from neutrons (accounting for effects of spectrum, dose-rate, limited penetration, injection of species, ...etc.). This links very much with the support offered by the modelling infratechnology (section 2.3.2), as well as with the accelerated testing envisaged for pre-normative research (section 2.3.1). Accelerated characterization would be best achieved by the creation and availability of suitable **integrated test-beds**, possibly including exposure and characterization in the same package.

An even stronger link with the modelling comes from the ambition to target the development of materials solutions based on **calculations**, for instance, by identifying in advance a range of promising material compositions or architectures for the relevant application. Whether these are reliable or not will depend enormously on the level of reliability achieved by the **ICME approach** (described in section 2.3.2) as well as, importantly, on which properties are of interest, since in many cases the prediction by calculation of mechanical properties may not be reliable, or even not possible. Here, too, **AI/ML** may be of use, depending on whether a sufficiently large amount of experimental data becomes available, by allowing extrapolation or interpolation to identify the best materials features to maximise a certain measurable property. HPC is here also an important asset.

The combination, possibly in an automated way, of accelerated screening with the use of calculations corresponds to the use of **high throughput approaches** for materials development, particularly if integrated fabrication methods (for instance additive manufacturing) are included.[54] In practice, this is unlikely to be immediately applicable in the case of the development of nuclear materials, at least in the near future, because of the complexity of the combined exposure to irradiation, temperature and environmental aggression, the size of the components involved and the high reliability required for long exposure times. Moreover, these approaches do not eliminate the requirements to license a material solution for a nuclear installation that are associated with the strictness of the safety authorities. It is, however, worth considering these new approaches, which are emerging in the field of materials science at large.

The relevant infratechnology here described corresponds to the third line in the warp of Figure 4, defined as "design, manufacturing and processing of innovative and better performing

[54] E.g. Chen, High-Throughput Computing for Accelerated Materials Discovery, in: Computational Materials System Design, Shin and Saal Editors, Springer (2018).

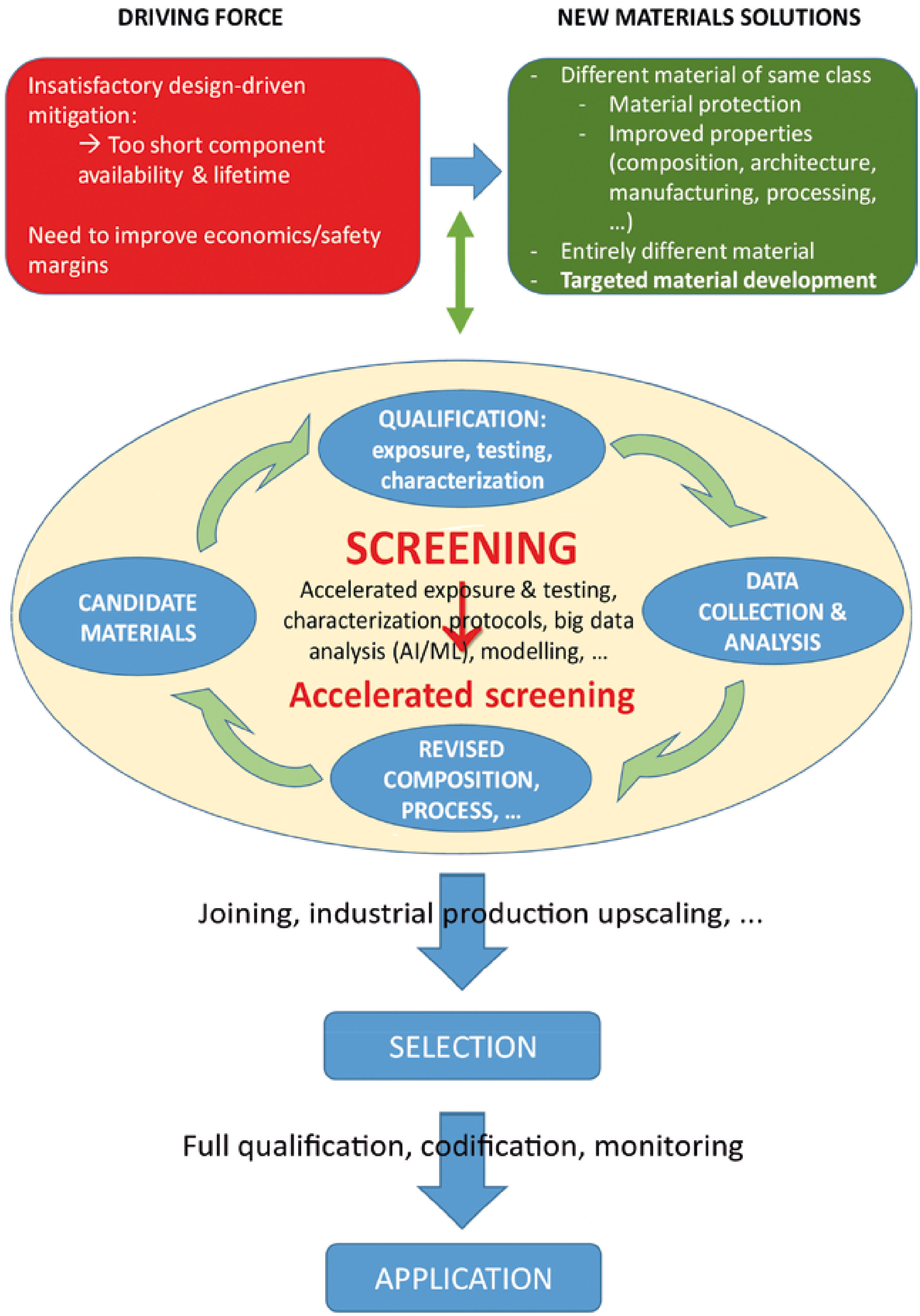


FIGURE 7: *Illustration of the process of development of new materials solutions and its desirable acceleration.*

materials", which can be concisely referred to as **advanced materials.** Figure 7 illustrates the procedure for materials development and its desirable acceleration.

2.4. *Data Management*

Trends in the research sector, such as the adoption of data management policies by funding agencies[55,56], publishing houses[57] and partner organizations[58], the emergence of data journals[59,60] and frameworks for data citation[61], are indicative that data management is becoming an integral component of the mainstream research process. While the specific reasons for this circumstance are varied, the underlying motivations are improved science and greater opportunities for innovation. Self-evidently, membership organizations that rely on data to achieve their objectives stand to benefit from improved data management practices. This is certainly the case for the EERA JPNM, where large volumes of materials' testing, characterisation and modelling data are generated that are of inherently high intellectual and/or commercial value.

In the case of EERA JPNM, the lack of data is a significant barrier to the formulation of design rules to be put in design codes and in feeding fuel performance codes, whether the data are needed to trace empirical curves or to validate/calibrate models. Historically, large amounts of test and measurement data have been generated through national and international research programmes, but these data are often not available or, if available, they are not complete (for instance data for time to creep-rupture, but not the creep-curve itself). While appropriate web-enabled databases have been developed, e.g. the materials database MATDB of Online Data & Information Network of the European Commission Joint Research Centre[62] or the International Fuel Performance Experiments database held by the OECD,[63] often data are not made available because they are protected by confidentiality and therefore cannot be shared, or in other cases because they were not properly stored. Furthermore, in ongoing and future research programmes, a very large share of the effort, and the cost, will be dedicated to material testing and property measurement, including novel materials. Since experimental data provide the basis for design curves and rules, as well as for the development and validation/calibration of models, their availability must be guaranteed with a view to their re-assessment in the context of new models, different operating conditions, different regulatory requirements, etc. Given that such data are also expensive to generate, it should become standard practice for the data to be collected, preserved, and made readily accessible, with respect to intellectual property rights, of course. The latter issue may require agreements, e.g. in terms of embargo periods, before the data become disclosed.

With improved data management depending on so many (sometimes competing) factors, best practices are evolving that are designed to support improved data management; these include data management plans[64,65] and the

[55] Data management - H2020 Online Manual. Retrieved 29 May 2018 from http://ec.europa.eu/research/participants/docs/h2020-funding-guide/cross-cutting-issues/open-access-data-management/data-management_en.htm

[56] Funders' data plan requirements. Retrieved 29 May 2018 from http://www.dcc.ac.uk/resources/data-management-plans/funders-requirements

[57] Availability of data & materials: authors & referees @ npg. Retrieved 29 May 2018 from https://www.nature.com/authors/policies/availability.html

[58] ILL Neutrons for Society - Data Management. Retrieved 29 May 2018 from https://www.ill.eu/users/user-guide/after-your-experiment/data-management

[59] Data in Brief - Journal - Elsevier. Retrieved 29 May 2018 from https://www.journals.elsevier.com/data-in-brief.

[60] Scientific Data. Retrieved 29 May 2018 from https://www.nature.com/sdata

[61] Welcome to DataCite. Retrieved 29 May 2018 from https://datacite.org

[62] https://odin.jrc.ec.europa.eu/odin/index.jsp

[63] https://www.oecd-nea.org/science/wprs/fuel/ifpelst.html

[64] DMPonline | Digital Curation Centre. Retrieved 29 May 2018 from http://www.dcc.ac.uk/dmponline

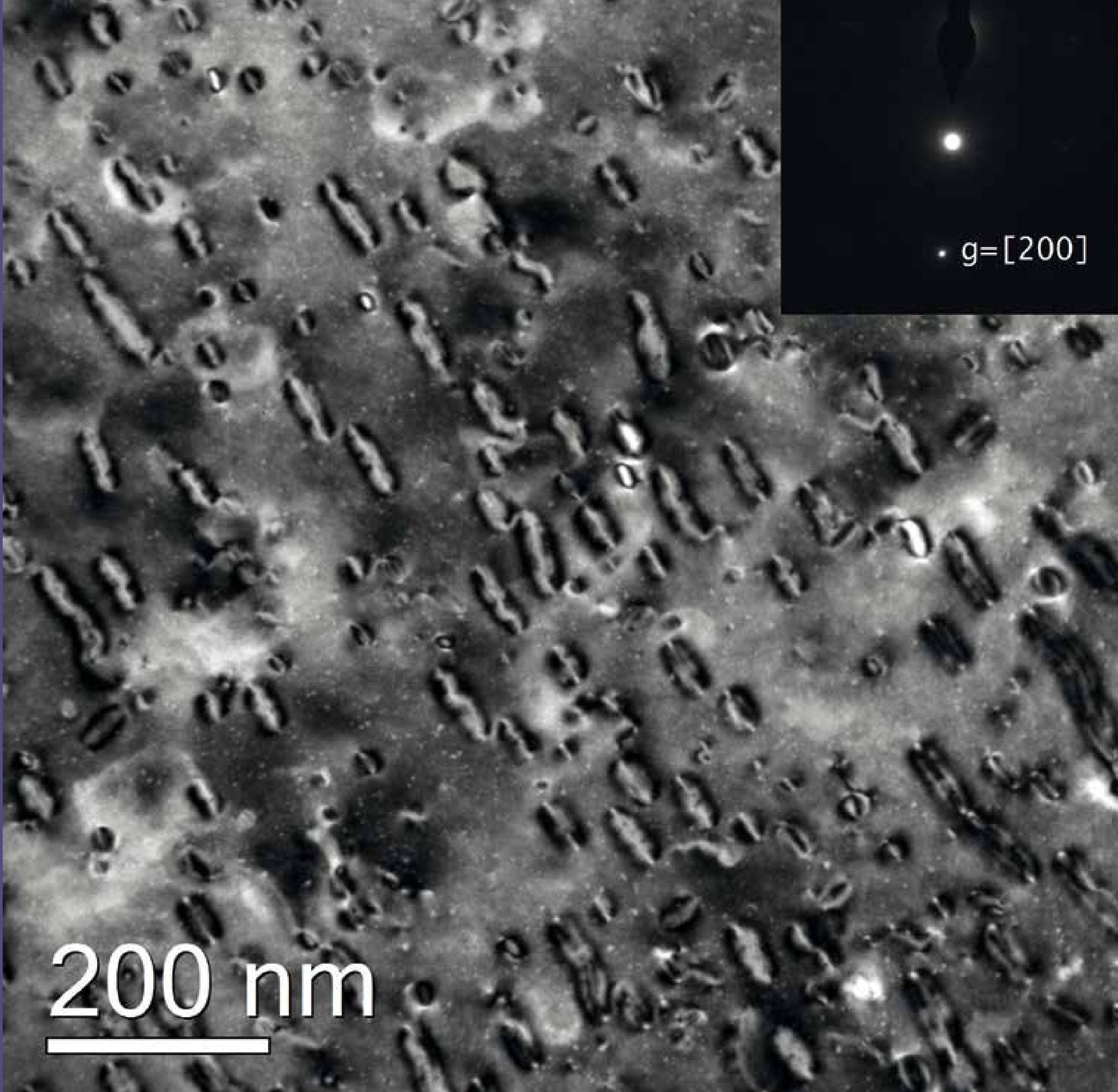
[65] EUROPEAN COMMISSION Directorate-General for Research & Innovation, Guidelines on FAIR Data Management in Horizon 2020 (2016). Retrieved 29 May 2018 from http://ec.europa.eu/research/participants/data/ref/h2020/grants_manual/hi/oa_pilot/h2020-hi-oa-data-mgt_en.pdf

FAIR (Findable, Accessible Interoperable and Re-usable) Guiding Principles for scientific data management and stewardship[66]. The FAIR guiding principles are concerned with data becoming findable, accessible, interoperable and reusable, so probably the most pressing need in the engineering materials sector is for **data interoperability**. In this context, standards for engineering materials data are needed to ensure consistency of the data coming from different laboratories and to facilitate transfer between different information systems. **There is therefore a urgent need to develop standard formats for engineering materials data and to leverage the highly interconnected information and communication technology (ICT) infrastructures that have emerged in recent years, with a view to ensuring that the data on which design curves and rules are based remain available for the purposes of future re-assessment.** It should also become established practice for projects to maintain a data management plan beyond the project duration, so that the resources for data management can be allocated and concrete deliverables identified.

[66] Wilkinson et al., The FAIR Guiding Principles for scientific data management and stewardship (2016), Scientific Data, 3, doi:10.1038/sdata.2016.18.

3

NUCLEAR MATERIALS RESEARCH AGENDA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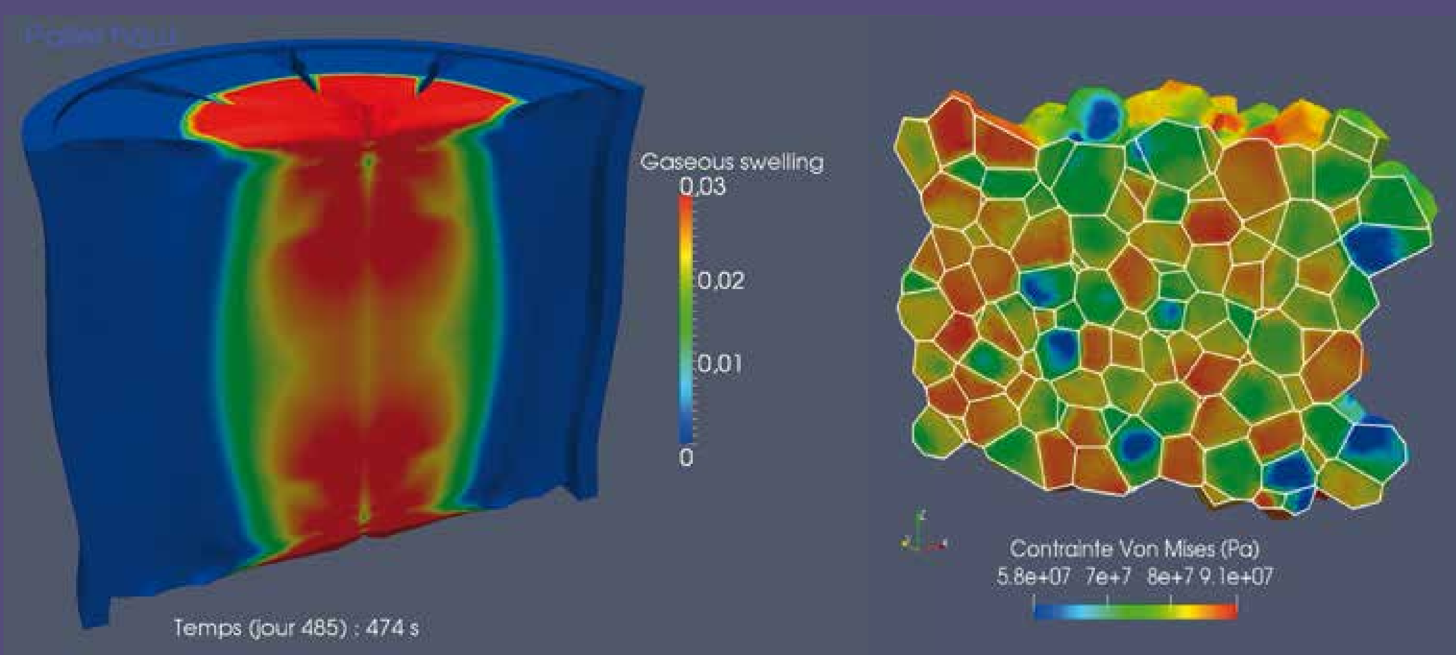
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Contrainte Von Mises (Pa)
5.8e+07 7e+7 8e+7 9.1e+07

3. NUCLEAR MATERIALS RESEARCH AGENDA

This section describes the materials research issues that need to be addressed to provide system designers and component manufacturers with suitable qualified materials and relevant design rules/fuel performance codes for the licensing and construction of GenIV reactor demonstrators, FOAK prototypes and eventually commercially plants. The two main subsections concern structural (3.1) and fuel (3.2) materials, i.e. the 'weft' of the 'fabric' of the SRA as in Figure 4. In each subsection the issues are organised according to the three complementary approaches illustrated in Figure 2 and described in section 2.3. namely: pre-normative materials research and qualification, advanced materials modelling and characterisation, and development and screening of advanced materials solutions. These correspond to the 'warp' of the SRA 'fabric' in Figure 4.

3.1. *Structural materials*

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

This section concerns the development of the assessment tools and the collection of the structural materials data in support of the design, licensing, and construction of the ESNII demonstrators, with a view also towards the FOAK prototypes and commercial reactors to be deployed later.[67]

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**.[68] All relevant slow long-term degradation processes need to be accounted for, especially **high-temperature processes** (creep, fatigue, thermal ageing), but also corrosion and low-dose-rate long-term irradiation effects. A fundamental issue is **how to measure material properties representative of 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

[67] Pre-normative research has been a key objective in the FP7 projects MATTER and MATISSE (http://www.fp7-matisse.eu/), and is the key focus in the H2020 project GEMMA (http://www.eera-jpnm.eu/gemma/) and in ongoing JPNM Pilot Projects. See in particular the summary report from the MATTER project for an overview of work done and envisaged: "Deliverable 9.1: The relevance of MATTER results for the design of ESNII reactors", http://www.eera-jpnm.eu/gemma/

[68] Technology Roadmap Update for Generation IV Nuclear Energy Systems, January 2014, https://www.gen-4.org/gif/upload/docs/application/pdf/2014-03/gif-tru2014.pdf

codes and standards. For instance, for creep an extrapolation by a time factor 3 is considered feasible, but this would require tests of 20 years duration for reliable 60 years design data. In a future low-carbon energy mix, the nuclear contribution will also 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 commercial deployment, and is thus a long-term need, there is a short-term R&D need to start in due time, i.e. now, in view of long-term test programmes[69].

Specifically, for the HLM-cooled systems, the most important issue is to demonstrate the structural integrity of components in prolonged contact with a corrosive coolant. There are no dedicated design rules, assessment procedures or material data in the codes to address the HLM environmental effects. Thus, there is an **urgent need** to perform **pre-normative research on the compatibility between components and HLM coolants** and develop procedures to be included in codes and standards.[70]

Importantly, most structural integrity issues in metallic components occur at the welds, the assessment of which is especially complex. This means that **much emphasis in the pre-normative research should be dedicated to welded components**.

Finally, the high investment cost is perhaps the largest obstacle for the development and deployment of innovative nuclear reactors. Thus, **in addition to demonstration of safety, the codes and standards should also ensure cost efficiency**. Pre-normative research related to reducing 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.

The **pre-normative research includes closely integrated experiments and modelling work at the meso- and macroscale**. The engineering scale is the basis for reactor design, but the understanding of the degradation mechanisms and of how to include them in the design rules and assessment procedures requires the proper length and time scale to be addressed for the particular degradation mechanism. One example is the extrapolation of accelerated data to operational conditions where the proper relevant degradation mechanism must be explicitly taken into account. The end product would then be engineering tools based on a multi-scale *informed* approach. This section thus includes engineering modelling approaches, while more mechanistic approaches are the subject of section 3.2.1.

In what follows the main pre-normative research issues for the key materials of the ESNII demonstrators are listed and summarised. More details can be found in the Technical Annex to the SRA, in section TA.1.1.1, which is a more extended version of this one.[71] At the end of this section, Table 3 (page 58) further summarizes the main pre-normative research issues. The 60-years design lifetime is not mentioned anywhere as a specific issue, because it corresponds to a combination of several of the issues listed in what follows and in the table. The **reference structural**

[69] See Final Report EERA JPNM Task Force 60 years Operational Life *Future Reactors Review and Roadmap for Future Activities*, available from http://www.eera-jpnm.eu/filesharer/documents/ and Technical Area 1a of the MoU between EERA and SNETP: *Development of a methodology for design and plant life management for 60-years operational life for non-replaceable components of Generation IV reactors*

[70] See Technical Area 1b of the MoU between EERA and SNETP: *Development of Design Rules, qualification and mitigation strategies for reactor components in heavy liquid metal coolants.*

[71] http://www.eera-jpnm.eu/?q=jpnm&sq=nboard

materials are austenitic steels, on which most research effort is focused, in particular 316L and 316L(N), that are planned to be used **for the core of all prototypes**. **In the case of the GFR, Ni-base alloys are also reference materials**, in particular the high-Ni alloy 800H for out-of-core components. Alloy 800H is qualified up to 750°C for the steam generator; but 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.[72] **Some emphasis remains on 9Cr F/M steels** which, although not used for MYRRHA and ALFRED, and only to a limited extent considered in ASTRID, remain **candidate materials for the phase II of some demonstrators**. For **cladding** the reference material is the **15-15Ti austenitic steel**.

High temperature behaviour and degradation of metals

Creep effects need to be taken into account above the negligible creep temperature, which is given in RCC-MRx for codified materials. Many fast reactor components operate in the creep regime, where **creep, creep-fatigue** and **thermal ageing** are the life-limiting factors.[73] In addition to the 60-years design life, high-temperatures must also be considered for accidental conditions, well-above the normal operational temperature. Accordingly, the following issues need to be addressed, focusing in particular on the **identification of mechanisms and development of relevant engineering models**:

- <u>creep and creep-fatigue deformation</u>: collect data and develop models that incorporate the dominant degradation mechanisms (e.g. non-isothermal creep-fatigue, especially serious in F/M steels that undergo softening under cyclic loading) or microstructural evolution at the proper scale;
- <u>creep and fatigue damage and crack propagation</u>: develop models that incorporate the initiation of damage and crack propagation until final failure and integrate creep and fatigue with proper damage criteria;
- <u>thermal ageing</u>: include thermal ageing effects in design codes and assessment procedures predicting the microstructural evolution with thermodynamic codes;
- <u>interaction creep, creep-fatigue and thermal ageing and upscaling</u>: develop models that combine effect of creep, fatigue and ageing for total life assessments of components, upscaling from mechanistic to polycrystal continuum models through homogenization schemes;
- <u>representative data collection and production for 60 years design life</u>: (i) compilation and assessment of 'historical' data, (ii) mechanical testing and microstructural analysis of materials from reactor components operated in the past, (iii) new test programmes.

Environmental compatibility between coolant and structural materials

<u>HLM-cooled systems</u>: In these systems the **main safety issue** is to guarantee integrity of materials in contact with the coolant for the lifetime of the component. It is thus necessary: (i) to address **corrosion**, **dissolution**, **erosion** by **mapping relevant rates as functions of all the variables involved**; (ii) address **liquid metal embrittlement** (LME) by identifying appropriate steps to be taken towards the **definition of relevant design rules** (either demonstration of immunity, or identification of mitigation strategies, or else complete mapping and use of coated or alumina-forming materials). The resistance to HLM degradation of **austenitic steels** needs further demonstration: **long terms tests** are needed as well as **accelerated tests in more aggressive environment**, as a complementary shortcut, supported by appropriate **models and microstructural**

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

[73] For a general overview of high-temperature assessment approaches see *Comprehensive Structural Integrity, Fracture of Materials from Nano to Macro Vol. 4 Cyclic Loading and Fatigue and Vol. 5 Creep and High-Temperature Failure*, 2007, Elsevier-Pergamon. For nuclear applications see *Structural Materials for Generation IV Nuclear Reactors*, Edited by Pascal Yvon, Woodhead Publishing Series in Energy, Elsevier, 2017.

TABLE 3: *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)	Experiments: For *long-term operation*: Mechanical tests of in-service material, long-term tests, accelerated tests. Basic tests for *model calibration*: creep, low-cycle fatigue, crack propagation tests. Special emphasis on long hold times. Microstructural analysis to link mechanism-based models to experiments are needed. Models: For creep, relaxation and cyclic deformation, emphasis on unified visco-plastic continuum models, mechanistic models for different creep mechanism and damage crack propagation fracture mechanics Models need to be translated into Design Rules or Assessment Procedures.
	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 and 15-15Ti with alumina surface protection	Mechanical tests: slow-strain rate tensile; fracture, fatigue, and creep-fatigue in flowing and stagnant conditions; Corrosion tests: Erosion and corrosion (oxidation and solution tests) in flowing conditions Qualification tests (mechanical and corrosion) for 316SS and welded components. Accelerated tests to map bounding conditions. Emphasis on long-term tests; A very careful documented control and monitoring of the test conditions (in particular oxygen control) is required for all tests. Tests to be complemented by detailed microstructural analysis (e.g. SEM, EBSD, XRD, TEM); Engineering related approaches need to be developed Coupling with models developed in 3.1.2
	Liquid Metal Embrittlement in HLM (LME)		
	Compatibility with HT He	Austenitic steels 316L and alloy 800, Ni-based alloys (Inconel 617, Haynes 230, Hastalloy-XR), as well as SiC/SiC and AFA steels.	
Radiation effects	Low temperature embrittlement & plastic flow localisation	Austenitic and F/M steels (irreplaceable & structural components)	Exposure to irradiation, also including coolant environment. Standard mechanical test in hot cells of neutron irradiated materials (irradiated in test reactors or in-service exposed) and complementary ion/proton irradiation. These tests need to be supported by irradiation models from 3.1.2 Transfer the data into reduction factors for material properties.
	Long-term/low dose irradiation in environment		
	High dose irradiation swelling and creep	Fuel cladding materials: 15-15-Ti austenitic steel	

<table>
<tr><th>Main issue</th><th>Breakdown in sub-issues</th><th>Materials concerned</th><th>Techniques/Methods</th></tr>
<tr><td rowspan="3">Assessment of complex loadings</td><td>Non-isothermal thermo-mechanical loads</td><td rowspan="3">All</td><td rowspan="3">Experiments:
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.
Modelling:
Finite element models of complex load cases and simplification to translate into design rule load cases.</td></tr>
<tr><td>Complex stress distributions</td></tr>
<tr><td>Load transients and beyond design conditions</td></tr>
<tr><td rowspan="4">Integrity and qualification of weldments and welded components</td><td>Residual stresses</td><td rowspan="4">All welds:
austenitic, F/M; Ni-based alloys and dissimilar metal welds</td><td rowspan="4">Experiments:
Residual stress measurements by neutron diffraction and simpler but less accurate methods such as X-ray diffraction, contour method.
Standard test for welded specimens (tensile, fracture toughness tests, fatigue)
Characterization of the different regions of a weld (small punch, nano and micro indentation)
Mock-up test of welded component for validation.
Microstructrual analysis (SEM, TEM, XRD, EDX)
Modelling:
Simulation of weld process and post-weld heat treatment for residual stresses;
Structural integrity assessment (defect assessment crack propagation, damage) of welded specimens and components by FEM.
Translation of structural analysis assessment into Design Rules.</td></tr>
<tr><td>Weld procedures</td></tr>
<tr><td>Degradation modes and defect assessment</td></tr>
<tr><td>Compatibility with HLM</td></tr>
<tr><td rowspan="4">Sub-size and miniature specimen test standardization</td><td>Sub-sized/ miniaturised specimens for mechanical property</td><td rowspan="2">Fuel cladding material, 15-15Ti</td><td rowspan="4">Experiments:
Various fuel cladding tests (internal pressure, ring-compression, small-punch, cone 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)</td></tr>
<tr><td>Thin-walled cladding tubes</td></tr>
<tr><td>Small Punch test</td><td>All</td></tr>
<tr><td>Micro-pillar tests and nano-indentation</td><td>All</td></tr>
<tr><td>Component and material health monitoring</td><td>- - -</td><td>All</td><td>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</td></tr>
</table>

characterization. LME is a serious issue in, especially, F/M steels, that lacks a complete understanding: engineering models exist but need to account for the microstructure.

Gas-cooled systems: the corrosion behaviour depends here to a large extent on the relative **impurity level in the coolant** and whether **oxidation**, **carburization** or **de-carburization** occur, so mapping of rates is needed in this case as well. Due to the high temperature of operation, the possibility of **environment and high stress synergism** on corrosion and crack formation, including **coupling with creep and creep-fatigue**, needs investigation. The large number of possible materials for GFR imposes a need for wide **screening**. High Ni alloys require particular effort of full codification.

Radiation 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. For **structural components** there are two border lines: the **negligible irradiation curve** that defines conditions for absence of significant irradiation effects, and the **maximum allowable irradiation** that is acceptable but needs irradiation effects to be accounted for. For 316L(N) **austenitic steels** these values are defined based on irradiation hardening and embrittlement. The **60-years operational life low dose may have significant effects** from helium production induced by transmutation and potential **He embrittlement**, due to the elevated operation temperature. For **fuel claddings** the main concern is **irradiation creep and swelling**, the latter also related to helium production, which are less problematic in F/M than in austenitic steels. However, in the case of **F/M steels**, **low temperature** (<350°C) **radiation embrittlement** may lead to a significant shift of the ductile-brittle transition temperature (DBTT), even after fractions of one dpa, and potentially to severe **loss of elongation due to plastic flow localisation**, which still needs to be properly accounted for in design rules.

For all materials, the possibility of **synergistic effects between irradiation and corrosion** by dissolution **and LME** need to be quantified.

Assessment of complex loadings

All structural components in a nuclear reactor are exposed to complex loadings. Good **descriptions 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**.

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 integrated part of pre-normative research programmes for GenIV conditions: (i) **defect assessment** through **inspection** can reduce costs and increase safety, (ii) **post-weld heat treatment** to minimize residual stresses in welds manufactured in accordance with code requirements is especially important, (iii) reliable methods to **measure and calculate residual stresses** need to be pursued, (iv) **mechanistic understanding** remains essential.

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[74] become essential for:

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

[74] Hyde et al., Requirement for and use of miniature test specimens to provide mechanical and creep properties off materials: a review, *Int.* Materials Reviews 52 (2007) 213.

- Very thin material layers 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.

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. The standardization work must go through the involvement of national and international standard committees with participation of main testing laboratories and designers. These activities should concern:

- miniaturised specimens for standard mechanical property assessment: these are smaller versions of already used and standardized specimens;
- thin-walled cladding tube tests: these tests require specific equipment and standardization;
- micro- and nano-mechanical testing methods: these are unconventional and/or truly miniaturised specimens/techniques, e.g. nano-indentation, compression/tension/bending tests on micropillars, etc.

Component and material health monitoring

Non-destructive examination (NDE) of components is crucial to verify, during **inspections,** that **components are performing according to expectations** and to **detect cracks** well **ahead of time**, as they could potentially 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 conventional approach is thus to monitor the material degradation by analysing surveillance specimen exposed to in-service conditions by mechanical tests and, if required, microstructural analysis. The current industrial tendency is, in contrast, towards **online materials monitoring and lifetime estimation based on NDE**, possibly by applying artificial intelligence-based big data analysis to derive patterns capable of warning sufficiently in advance of the possibility that the component may fail.

3.1.2. Advanced structural materials modelling and characterization

The issues to be addressed with an ICME approach (section 2.3.2) are necessarily strongly linked to those addressed in pre-normative research (previous section), as they correspond to the main degradation processes that affect structural materials for GenIV systems. Since the early 2000s, 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,[75] FP7/GETMAT,[76] FP7/PERFORM60,[77] FP7/MatISSE.[78] Some of the relevant activities currently continue in the H2020/SOTERIA[79] and the H2020/M4F[80] projects. Issues connected with the modelling of the high temperature behaviour of F/M steels have also been partially addressed in FP7/MATTER[81] and FP7/MatISSE,[77] while the

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

[76] http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/46/040/46040904.pdf?r=1

[77] 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

[78] www.fp7-matisse.eu; http://www.fp7-matisse.eu/wp-content/uploads/2015/12/MatISSE-2015-GEN-IV-materials-Holmstrom.pdf

[79] www.soteria-project.eu

[80] http://www.h2020-m4f.eu/

[81] http://www.eera-jpnm.eu/?q=jpnm&sq=nboard; http://proceedings.asmedigitalcollection.asme.org/proceeding.aspx?articleid=1627574

modelling of oxidation and dissolution, as well as prolonged irradiation, in austenitic steels is being partly addressed in H2020/GEMMA[82]. Several modelling activities are also 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 not only still a lot of room for the improvement of existing models and there are still several open issues to be addressed, but there are in fact cases where no defined computer-simulation based physical modelling approach has been identified yet, beyond classical continuum approaches.

The most important issues to be addressed are briefly described in what follows and summarised in Table 4: Main issues concerning structural materials advanced modelling and characterization (page 64).

A more detailed account is provided in section TA.1.1.2 of the Technical Annex.[71]

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, which 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-orientated experiments** (see section 2.3.2). Modelling the **microstructural and microchemical evolution under irradiation** and/or while **exposed to high temperature** is essential in order to understand virtually all degradation processes.[83] Although significant advances have been made in the development of microstructural and microchemical evolution models in the last couple of decades, multiple challenges remain to be addressed in order to develop reliable ICME approaches 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)**.

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, as well as to bridge between scales.[84]

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. Issues to be addressed are **radiation hardening and embrittlement**, **plastic flow localisation** and deformation at constant load due to **irradiation 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 for crystal plasticity**[85] and devise

82 http://www.eera-jpnm.eu/gemma/

83 A summary of key irradiation effects in structural materials can be found in, e.g.: Zinkle, Phys. Plasma. 12 (2005) 058101; Zinkle and Was, Acta Mater. 61 (2013) 735.

84 Castin et al. Comp.Mater. Sci. 148 (2018) 116.

85 Volegov et al., Physical Mesomechanics 20 (2017) 174 https://link.springer.com/article/10.1134/S1029959917020072.

homogenization schemes[86] for continuum mechanics approaches applied to components. The challenge 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 effective because both microstructures evolve simultaneously over time.

Mechanical behaviour at high temperature

The model developments sketched in the section 2.3.2 are potentially applicable also in the case of high temperature mechanical behaviour, mainly describable as **thermal creep**, which may be 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. These models are currently not developed to the same level of advancement as microstructure evolution or plasticity models. It is especially important to allow for changes of creep mechanisms between accelerated laboratory tests and actual operation conditions. Moreover, for many engineering applications connected with high temperature operation in particular, thus affecting crucially GenIV systems, **finite element models may not be usable**. This happens when there is no sufficient understanding of the underlying processes available to guide the development of a consistent integrated modelling approach. In these cases, microstructural examination together with the semi-empirical identification of the important variables are essential.

Fracture mechanics

All materials contain defects, i.e. microcracks from which 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, **non-destructive examination techniques are key**, because currently **neither plastic nor visco-plastic models** can explicitly describe **crack initiation**. Thus, developments in this direction are needed, assisted by extensive dedicated microstructural examination. **Crack propagation** can be described under simple loading conditions but not in the case of complex loading, such as for instance under creep-fatigue. **Local** modelling **approaches** are to be pursued.

Coolant compatibility models

The processes that govern the **interaction between solids and coolants** are crucial in the context of GenIV reactors and need to be appropriately modelled beyond current capabilities.[87, 88]

More detailed approaches are required to understand the origin of the processes observed through microstructural examination. A full description of these processes requires the **identification of new appropriate modelling approaches**, because the complexity of the relevant physical and above all chemical processes challenges the possibilities offered by existing simulation tools. Especially challenging is the problem of **liquid metal embrittlement**, which requires modelling of the liquid/solid interaction, which is assumed to occur by adsorption of embrittling atoms from the liquid at stress concentrators along interphases, and the induced embrittlement and fracture mechanisms. Depending on the specific solid/liquid couple, fracture can be intergranular or transgranular and although it

86 Allen, Comp. Sci. & Technol. 61 (2001) 2223.

87 Zhang et al., Review - Models of liquid metal corrosion, J. Nucl. Mater. 404 (2010) 82.

88 Zhang, A review of steel corrosion by liquid lead and lead-bismuth, Corrosion Science 51 (2009) 1207.

TABLE 4: *Main issues concerning advanced structural materials modelling and characterization.*

<table>
<tr><th>Main issue</th><th>Breakdown in sub-issues</th><th>Main materials concerned</th><th>Techniques/Methods</th></tr>
<tr><td rowspan="7">Microstructural and microchemical evolution</td><td>Formation of radiation hardening and embrittling microstructural features (low temperature)</td><td rowspan="5">F/M (possibly also ODS) and austenitic steels, ceramics</td><td rowspan="7">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.
Machine learning / artificial intelligence.
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.</td></tr>
<tr><td>Formation of voids and onset of swelling (high temperature/dose)</td></tr>
<tr><td>Radiation-induced segregation/precipitation at extended defects</td></tr>
<tr><td>Microstructural evolution under load in relation with irradiation creep</td></tr>
<tr><td>Thermal ageing and subsequent precipitation/ segregation</td></tr>
<tr><td>Oxide formation/ dissolution in ODS during fabrication, thermal ageing and irradiation</td><td>ODS alloys</td></tr>
<tr><td>Correct interpretation of microstructural examination technique signals and raw data</td><td>Any</td></tr>
<tr><td rowspan="3">Mechanical behaviour after and under irradiation (low temperature)</td><td>Radiation hardening: flow behaviour</td><td rowspan="3">F/M and austenitic steels</td><td rowspan="3">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.</td></tr>
<tr><td>Plastic flow localisation</td></tr>
<tr><td>Irradiation creep</td></tr>
<tr><td rowspan="3">Mechanical behaviour at high temperature:</td><td>Cyclic plasticity (softening and hardening)</td><td rowspan="3">Austenitic and F/M steels</td><td rowspan="3">Different models at different length-scales are needed: from dislocation based to continuum mechanics. The techniques and methods are as in the previous case, but their use is more challenging because of the number and complexity of possible mechanisms, involving not dislocations and grain boundaries, and complex thermo-mechanical loading conditions. The up-scaling includes crystal plasticity and visco-plastic models.</td></tr>
<tr><td>Thermal creep</td></tr>
<tr><td>Creep-fatigue interaction</td></tr>
<tr><td rowspan="2">Fracture mechanics</td><td>Crack initiation</td><td rowspan="2">Austenitic and F/M steels</td><td rowspan="2">(Visco)-plastic models including damage criteria; classical fracture mechanics governed by crack tip parameters; local approaches where the fracture processes are explicitly modelled</td></tr>
<tr><td>Crack propagation</td></tr>
</table>

Main issue	Breakdown in sub-issues	Main materials concerned	Techniques/Methods
Compatibility with coolants & coolant chemistry	High temperature oxidation/corrosion	F/M and austenitic steels	Atomistic and thermodynamic models for corrosion and LME; 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.

appears as cleavage some micro-plasticity is always involved. The detailed mechanisms are still largely unknown, or at least still debated.

Models in support of the use of charged particle irradiation

Neutron irradiations are long and expensive experiments and the access to the relevant facilities is generally restricted. Charged particle irradiation (ions, protons, electrons, …) is a very valuable and affordable tool for the purpose of modelling-orientated experiments and materials screening. However, transferability issues exist between charged particle and neutron irradiation environment (see also section 6.1.3).[89] It is thus **essential that suitable models address the specificities of these modes of irradiation**. Difficulties need also to be overcome concerning the post-irradiation examination, particularly the possibility of obtaining a meaningful assessment of mechanical properties after charged particle irradiation. Overall, the **goal** is to develop an established and possibly **standardized methodology**, based on **models**, to perform **charged particle irradiation and relevant PIE**, including **mechanical characterization**.

Development of standard methodology for model validation

The validation of models is a complex and costly task that requires specific attention also from the methodological point of view, in order to reduce uncertainties to the minimum and assess those that are associated with the model and cannot be avoided. **Modelling-orientated experiments** need to be designed on appropriate materials and trying to **separate variables**. **Protocols** need to be established to

[89] Was, J. Mater. Res. 30 (2015) 1158.

apply **microstructural characterization** techniques and analyse their outcome. Methods to **simulate** the response of materials to experimental techniques are needed. **Protocols** for the **comparison** between **simulation and experiment** need to be established, as well.

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, the development of microstructure evolution physical models for austenitic steels is much less mature than for F/M steels. In general models for concentrated alloys are less developed than for diluted alloys:

- **The modelling of F/M steels is worth being pursued** because these materials remain important in the long-term for GenIV applications, and especially because they represent an important reference case for the methodology; F/M steels represent also the starting point to develop models of use to simulate ODS and/or alumina-forming alloys (FeCrAl);

However:

- **The modelling of austenitic steels needs to be pushed forward**, "catching up" with the level of maturity that characterises models for F/M steels on the low-scale side.

Ceramic materials are often chemically and crystallographically complex systems (e.g. MAX phases), that require specific developments. In terms of atomic-level modelling, SiC and Al_2O_3 are probably the most studied, although the activities are globally very limited and scattered. Importantly, these materials are generally used in the form of composites, which opens different types of problems for modelling, one of them being the monitoring of the behaviour of the architecture, such as via 3D tomography, and the assessment of the role of the interfaces between e.g. fibres and matrix.

3.1.3. Development of advanced structural materials

The development and codification of new materials solutions 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, for example, the use of optimised materials for the short term ESNII prototypes. Yet, for further demonstrator phases, FOAK prototypes and longer term applications in commercial reactors it is important to pursue the development and codification of better performing materials and materials solutions.

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:

- Improved austenitic steels;
- Ferritic/martensitic steels;
- SiC_f/SiC composites;
- Refractory metallic alloys;
- Modified surface layers for protection against 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. In what follows, possible development paths for each class of materials are briefly listed. At the end, Table 5 (page 68) summarizes the main issues related with the different classes of advanced structural materials. More details can be found in section TA.1.1.3

of the Technical Annex, which is an extended version of the present one (next page).[71]

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 as a large database of irradiated materials is available. 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**, [90] 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% (ΔV/V, where V is the volume). The path is for development of **swelling resistant austenitic steels**, by further stabilising and optimising steels of the 15-15 Ti-class, mainly by working on the composition, with addition of swelling inhibitors like Ni, Si and P, as well as Nb and V.

Another issue affecting austenitic steels is that, despite their generally higher resistance to corrosion than e.g. F/M steels, they **may not offer sufficient guarantees of corrosion-resistance in HLM-cooled systems**,[91] especially with a view to pushing up the operating temperature for higher efficiency. Here one of the most promising paths currently pursued is the development of **alumina-forming austenitic (AFA) steels** that, through the addition of 4-6 %wt Al, exhibit superior oxidation resistance up to 900°C, due to the formation of a protective Al_2O_3 scale. They also offer creep strength comparable with some superalloys that contain a much higher amount of nickel, thanks to strengthening via precipitation of NiAl particles. Nevertheless their application as fuel cladding materials will require several compositional changes and qualification under neutron irradiation, before converging to a suitable composition. Moreover, the threat of LME remains open and will require the identification of a proper framework for materials codification and licensing.

Ferritic/martensitic steels

Despite their high thermal conductivity and excellent dimensional stability under neutron irradiation (swelling resistance), the core applications of F/M steels in fast reactors has been so far limited. Their **use as cladding or core material in other systems is at present prevented by**:

- Loss of strength at high temperature (T>550°C), including softening under cyclic operation;[92]
- Neutron irradiation embrittlement al low temperature (T<350°C), including plastic flow localisation with subsequent drastic reduction of uniform elongation;[93, 92]
- Susceptibility to liquid metal embrittlement in contact with HLM.[94]

However, 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**. Their use in future GenIV commercial power reactors will require a number of actions:

- Improvement of **high temperature mechanical behaviour**;
- Improvements of **ductility after low temperature irradiation**, including **slip localization** with **loss of uniform elongation**;
- Improvement of the **compatibility with HLM**, by investigation the mechanisms responsible for corrosion and, even more importantly, **liquid metal embrittlement**.

[90] C. Cawthorne, E.J. Fulton Nature, 216 (1967), pp. 575-576.

[91] K. Lambrinou, et al Journal of Nuclear Materials, Volume 490, 2017, pp. 9-27.

[92] Raj & Vijayalakshmi, Ferritic Steels and Advanced Ferritic-Martensitic Steels, in: Comprehensive Nuclear Materials 4.03 (2012) 97.

[93] Farrell, Byun and Hashimoto, J. Nucl. Mater. 335 (2004) 471.

[94] Ersoy et al., J. Nucl. Mater. 472 (2016) 171.

TABLE 5: *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	Multi-stabilized steels	Addition of elements (Ti,Si,P, Ni …) that (a) in solid solution enhance point-defect recombination (b) create nanoprecipitates that stabilize the dislocation network
	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.
	Welding and relevant qualification	ODS steels	Development, optimization, standardization of welding procedures for ODS steels
	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	All F/M starting from baseline steel	(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 thermodynamic modelling Realization of sample joints and prototypes Mechanical testing and performance assessment in the operative conditions Post-test examination by SEM, TEM, XRD
	Standardization of mechanical tests and analysis for tubular structures of continuous fiber ceramic composites		Standardization of testing procedures Development of standards & design rules

Main issue	Breakdown in sub-issues	Materials concerned	Techniques/ Methods
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)		Protection with coatings and relevant qualification.
	Characterise and improve behaviour towards low temperature radiation embrittlement		Improve DBTT shift under irradiation through composition and processing changes. Verification through neutron irradiation at relevant DPA doses, post irradiation characterization by TEM, SEM, mechanical testing.
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 prolonges 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, etc.) 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.

Two alternative, though not mutually excluding and partly complementary, paths are considered to **improve high temperature strength** of F/M steels, namely (i) **oxide-dispersion strengthening (ODS)** using powder metallurgy or alternative production processes, that can potentially reach **operating temperatures above 700°C**, offering also high radiation resistance,[95] and (ii) **creep-strength enhancement (CSE)**, through composition and thermomechanical treatment optimization, leading to optimised carbide distributions and microstructures, including the assessment of

[95] P. Dubuisson, Journal of Nuclear Materials, Volume 428, Issues 1–3, September 2012, pp. 6-12.

the potential cyclic softening, low temperature irradiation (slip localization) and compatibility with coolants.[96] The main issue affecting ODS cladding is 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, affected by **limited reproducibility** of the quality of the end product. Thus the **TRL of ODS steels remains low** and the priority is to **improve the production routes**, in partnership with steel-makers. There is also an open issue of appropriate **welding procedures**. CSE has the advantage that it involves conventionally produced (i.e. by casting in crucible) F/M steels, However, it is unlikely that operating temperatures higher than 650°C, as already achieved outside nuclear energy, can be targeted. No nuclear grade steel of this type currently exists. It is considered that appropriate compositional tuning and thermomechanical treatments may also help **minimise** the problem of **low temperature radiation embrittlement**.

The problem of **corrosion by dissolution and erosion** can be faced either by applying suitable **surface modification** (see below), or by producing **corrosion resistant steels** that produce such a protective layer by themselves, e.g. through **Al addition (FeCrAl)**.[97] These steels exist but still need composition optimisation, 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 and Al. It is possible - although it remains to be demonstrated - that these alumina-protected steels may also provide protection against LME: the still missing clear understanding of the mechanisms behind LME in F/M steels and the synergy between LME and irradiation hardening prevents clear conclusions to be drawn in this respect. The identification of a proper framework for materials codification and licensing whenever LME is involved is expected to be complex.

[96] R.L. Klueh, et al, Scripta Mater., 53 (2005), p. 275

[97] Jun Lim et al. Journal of Nuclear Materials, Volume 441, Issues 1–3, October 2013, Pages 650.

SiC_f/SiC composites and other ceramics

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. 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,[98] while being considered also for advanced accident tolerant fuel cladding for GenIII+ reactors.[99]

The **main weakness** of these composites as cladding materials is that, both as produced and especially after deformation, they are **not hermetic** and therefore **cannot retain the fission products**. Suitable solutions to this problem have, however, been proposed and are being tested.

There are four issues that require attention and further work for the full qualification of SiC_f/SiC composite cladding:

- behaviour under irradiation;
- corrosion issues;
- joining;
- standardization of testing procedures.

[98] A. K. Rivai, M. Takahashi, J. of Pow. and Ener. Sys. 1 (2007) 134; M. Takahashi, M. Kondo, Prog. in Nucl. Ener., 53 (7) (2011) 1061; M. Kondo, M. Takahashi, J. Nucl. Sci. and Tech 43(2) (2006) 174; A.K. Rivai, M. Takahashi, Prog. Nucl. Energy 50 (2008) 560; M. Takahashi, S. Uchida, Y. Kasahara, Prog. Nucl. Energy 50 (2008) 197.

[99] http://www.westinghousenuclear.com/Portals/0/Technovation%20Stuff/Accident%20Tolerant%20Fuel%20Brochure%20.pdf; http://www.ga.com/accident-tolerant-fuel.

SiC is also considered as material for **thermal shielding, insulation**, **control rods and seals**, for high-temperature gas-cooled reactors, together with other materials such as carbon composites, mullite, Al_2O_3, TiO_2, ZrC, ZrN, B4C, graphite or graphene.

SiC behaviour under irradiation

Nuclear-grade SiC_f/SiC composites have shown to be stable to extremely high irradiation doses. Overall swelling of monolithic SiC remains less than 1-2% in the whole range of possible irradiation temperatures, therefore the main issue to be addressed concerns swelling in the composite used for the actual component (fuel cladding tube). 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. 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 to their behaviour under neutron irradiation.

SiC corrosion Issues

Chemical compatibility in reactor environments is highly dependent on the thermodynamic stability of SiC in the coolants and in contact with the possible impurities contained therein. For example helium coolant may contain small amounts of gas impurities such as CO_2, CO, H_2O, H_2, CH_4, O_2, as well as solid particles coming from a variety of sources throughout the reactor system.[100] The key oxidising impurities to consider in the He coolant will be oxygen and water vapour: the **reaction of SiC with O_2 and moisture** at elevated temperatures leads to three typical oxidation features, passive oxidation, active oxidation and bubble formation. 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.

[100] Estimation for ALLEGRO from HTR-10 experiments.

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**. 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 as a pre-requisite to characterize the behaviour of the material in environment (tightness against fission products, contact with flowing He, irradiation). **Standards on mechanical tests for nuclear grade SiC_f/SiC are thus necessary** to produce accurate and reliable data, based on well-defined test methods, detailed specimen preparation, comprehensive reporting requirements, and commonly accepted terminology. For instance, the failure behaviour of SiC_f/SiC components having tubular geometries is anticipated to be significantly different from that observed for flat two-dimensional architectures.

Other ceramics

The design of the high temperature gas cooled reactor concept (GFR, V/HTR), imposes the need for materials solutions to be devised for the purposes of **thermal shielding and insulation**, as well as **control rods and seals**. These materials must be gas-tight, corrosion resistant, and exhibit high fracture and creep strength, while being inexpensive and preferably fabricated in a net-shape design, with the ability to be joined. Achieving reasonable leak tightness is one of the most important issues, since He gas permeates easily through most materials. While also some metallic materials are being explored, (Inconel/Ag) and ceramics are especially being investigated and tested for O-ring seals. In many cases the material chosen also should be resistant to shocks related to thermal transients, which might occur when the flow of the process fluid or the coolant is interrupted. Therefore, next to the super alloys, ceramics with improved fracture toughness are being developed, i.e. for intermediate heat exchangers. At the moment, in addition to SiC (deposited using CVI), also other materials such as carbon composites, mullite, Al_2O_3, TiO_2, ZrC, ZrN, B4C, graphite or graphene are being considered for this function.

Refractory alloys

The requirements of safe operation 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°C 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. However, these alloys exhibit **high affinity for oxygen, hydrogen, nitrogen and carbon**, which easily diffuse in the bulk causing **hardening and embrittlement** during both the production process and the service life. The upper temperature limit of applicability of these materials is mostly determined by oxygen corrosion and interstitial impurity embrittlement. The low temperature limits for their use under neutron irradiation are determined by irradiation embrittlement and DBTT shift at or above the operational temperatures.

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.

Several refractory metals and alloys have been considered in fusion for first wall and blanket structural materials applications. Here only two alloys of this type are further discussed as prominent examples, both studied to some extent in the past and recently experiencing renewed interest for cladding: V and Mo alloys.

Vanadium alloys

High quality manufacture of V-4Cr-4Ti heats of high purity, with state-of-the art properties required for fusion blanket application, has been demonstrated into a variety of engineering-relevant shapes, including small diameter thin wall tubes. Research on V-alloys as clad materials for the previous generation SFR was initiated in Europe during the 1970s but soon discontinued. V-alloys remain, however, on the list of potential interests for GenIV applications.

The essential condition to make V- alloys potentially suitable for core applications is to enlarge its reference operating window, by expanding both the low and high operation temperature limits. This involves finding 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 their high vulnerability to oxidizing species and proneness to embrittlement by impurities that are unavoidable in fast reactor coolants. The development of adequate protective barriers ensuring protection against oxidation and corrosion (see below) is probably mandatory. Attention should be focused on suitable fabrication and processing technologies that avoid contamination, are compliant with operative requirements and nuclear environment, and are applicable to complex surfaces, including tubes and their internals.

Molybdenum alloys

With a melting temperature in excess of 2600°C, molybdenum is the most versatile among the refractory metals and an excellent structural material. Mo alloys have been extensively investigated for their application as cladding of fast spectrum high temperature gas cooled reactors for space, as well as for fusion first wall applications and, more recently, ATF cladding. Sufficient experience on fabricability and joinability exists, so component construction is not a major concern. Molybdenum and its alloys have also good swelling behaviour under neutron irradiation, less than 3% up to 110 dpa.

The main issues are:

- The affinity for oxygen, carbon and nitrogen, which easily diffuse in the lattice occupying the interstitial sites, causing embrittlement;
- The susceptibility to oxidation corrosion in oxidizing environments where the low melting temperature (795°C) and the formation of volatile oxides may give rise to catastrophic oxidation,[101] although Mo has good corrosion resistance in several liquid metals;
- The limited ductility and irradiation induced embrittlement with, depending on the dose, DBTT shift up to 800°C for unalloyed Mo;

The improved ductility of the oxide dispersion strengthening makes molybdenum an interesting option for core applications. ODS Mo-La alloys show a DBTT at -100°C in the unirradiated state and an outstanding reduction in the DBTT shift after irradiation, even at low temperatures; although, depending on the mechanisms leading to the observed corrosion resistance, their application as cladding may require the deposition of a suitable diffusion barrier to prevent oxidation corrosion and mass transfer issues for high temperature applications. Systematic evaluation of these alloys is needed to verify whether they meet the challenging requirements for application to the core of the HLM cooled systems, via: (a) characterization of the corrosion in HLM of varying [O] and temperature; (b) characterization of the creep performance; (c) mechanical testing to investigate LME issues; (d) development of welding procedures; (e) development of barrier coatings to prevent oxygen gathering; and (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 and wear 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**. This need is common to (a) steels in HLM operating above 400-450°C; (b) SiC_f/SiC in oxygen deficient high temperature environments; (c) Refractory alloys, (d) parts and components subject to fretting wear (stellite –Co- replacement issue).

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

1. The SL should form thermodynamically stable passivating 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, via introduction of an interdiffusion barrier or substrate where the diffusion rate of the SL species is high.

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.

4. If deposited, the SL deposition processes should be carried out at low temperature to avoid the degradation of the substrate

[101] Simnad and Spilners JOM 7 (1955) 1011. https://doi.org/10.1007/BF03377603; Lai, High-Temperature Corrosion and Materials Applications, ASM Intl. Materials Park Ohio; Smolik et al., Oxidation and Volatilization of TZM Alloy in Air, 9th International Conference on Fusion Reactor Materials ICFRM-9 10 - 15 Oct 1999, Colorado Springs, CO, USA.

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 **GESA**;
- ceramic Al_2O_3 barrier coatings produced by **pulsed laser deposition** (PLD).

These two are discussed in the Technical Annex (section TA.1.1.3) as prominent examples,[71] together with **detonation gun spraying** as an especially straightforward solution, and **atomic layer deposition** (ALD) as an emerging technique. Other processes are available and possible, amongst other coatings based on refractory alloys, alumina forming alloys, silica forming alloys and MAX phases. FeCrAl coatings produced by pack cementation are considered for the steam generator of ALFRED.

The **qualification of coated materials** is especially delicate because, bearing 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 implies foreseeing not only tests aimed at verifying the stability of the protective SL and standard qualification procedures (See section TA1.1.1 of the Annex to this SRA, http://www.eera-jpnm.eu/?q=jpnm&sq=nboard), but also including in any case the qualification of the substrate material. This is necessary in order to know which corrosion-rate or effects in general should be expected in the case of SL failure, as well as, crucially, worst case condition testing, according to criteria that will necessarily depend on the individual case.

Welding coated components may also pose issues because of the likely risk that the protection is lost in the process of welding.

Prospective materials

The materials solutions listed in the previous sections correspond to 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 which for the moment have hardly ever been 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.[102] Elements are combined in roughly equimolar concentrations so that, in theory, the high entropy of mixing stabilizes simple solid-solution phases with relatively simple crystal structures. These alloys may be compositionally complex, but can be 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.[103] 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 key point is that their conception offers the possibility to tailor via a suitable alloy design the desired thermo-mechanical, corrosion and radiation resistance properties, by modifying the composition. The field offers therefore wide opportunities to explore, discover, and develop new classes of alloys for structural

[102] Yeh et al. Adv Eng Mater 6 (2004) 299; Huang et al. Adv Eng Mater 6(1-2) (2004) 74; Yeh Ann Chim Sci Mat 31(6) (2006) 633; Zhang et al. Adv Eng Mater 10 (6) (2008) 534.

[103] Liu et al. Transactions of Nonferrous Metals Society of China 25 (4) (2015) 1341.

and functional applications. Further research should be devoted to explore their applicability to the GenIV systems.

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.[104] They are versatile materials, whose properties can be tailored by forming solid solutions on the M, A and X sites, that often exhibit better properties than the 'parent'.[105] In particular, these materials are characterized by unusually high –for ceramics– damage tolerance. In terms of response to irradiation, MAX phases seem to have a remarkable capacity for self-annihilation of neutron-induced defects at elevated temperatures.[106] They are not, however, refractory materials, i.e. they will have a limitation in terms of operating temperature, dictated by their stability in the specific working environment.[107] Because of their excellent compatibility with HLM they are promising core materials for HLM-cooled systems. 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.
- Phase purity.
- Collection of statistically-relevant experimental data.

[104] Barsoum, MAX Phases: Properties of Machinable Ternary Carbides and Nitrides, 2013 Wiley-VCH Verlag GmbH & Co. KGaA, Weinheim, Germany.

[105] Lapauw et al., Inorganic Chemistry 55 (2016) 5445; Tunca et al., Inorganic Chemistry 56 (2017) 3489.

[106] Tallman et al. J. Nucl. Mater. 468 (2016) 1; *ibidem* 484 (2017) 120; Ang et al. Scripta Materialia 114 (2016) 74; Ang, et al., Journal of the European Ceramic Society 37 (2017) 2353.

[107] Low, Thermal Decomposition of MAX phases, https://www.azom.com/article.aspx?ArticleID=6711.

3.1.4. Summary of structural material 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 for each of them. This is done in Table 6 (next page).

For each of the many possible routes towards novel materials solutions that can be pursued, there are weaknesses that may act as show-stoppers. In principle, if the materials solution fails to pass the test concerning this issue, it may be sensible to abandon that route. This issue is broadly addressed in the Risk Assessment section (section 11).

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 particular for fuel performance codes (see section 2.3.1), in support of the design and licencing of fuels for ESNII prototypes and later GenIV reactors.

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

TABLE 6: *Summary of issues relevant to the different structural material classes and types.*

Materials	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		Improve swelling resistance and compatibility with coolants (apply high temperature protective barriers).
	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 (not suitable for HLM coolants due to corrosion issues).	To identify mitigation strategies, models of radiation-induced microstructure evolution in connection with predictions of embrittlement and swelling (He production). Creep and creep-fatigue engineering models in support of design correlations and rules.	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. Not suitable for irradiation field environments due to swelling and embrittlement. Also not suitable for HLM coolants due to corrosion issues.		Compatibility with coolant at high temperatures; manufacturing and joining

Materials \ Type of related issues		Pre-normative research	Modelling	Advanced materials' solutions
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"

Materials \ Type of related issues		Pre-normative research	Modelling	Advanced materials' solutions
Ceramics (cladding and coating)	Non-metallic core support structures (ad hoc ceramics)	Screening of candidates. Test standardization (mechanical & thermophysical properties)	Microstructure evolution models under irradiation	Protection against oxidation
	Al_2O_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, though stability to high temperature needs to be verified case by case. Need extensive investigation for screening, including understanding of origin of properties, before applications are identified. Usable as coatings.		

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 above-mentioned fuel parameters. . If the phase diagram of the (U-Pu-O) system is largely known[108], further data is needed at high temperature and in presence of minor actinides or fission products, which lead to significant changes in fuel composition. Data on thermal conductivity and specific heat are even scarcer.

Atom transport and microstructural evolution

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. There is very limited literature information on diffusion in mixed oxide fast reactor fuel published since 1990.

Fission Product and Helium behaviour

The cladding constitutes the second retention barrier against radioactive fission product release.[109] A design criterion is the absence or

[108] C. Guéneau et al. J. Nucl. Mater. 419, 145 (2011).

[109] The first retention barrier is the inherent retention capability of the fuel. Attempts at improving this exist both for UO2

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. Quite extensive information behaviour or rare gases is available on fresh and irradiated UO_2 and much progress has been made in the last two decades in the understanding of the mechanisms governing the behaviour in this compound.[110, 111, 112] The same progress must now be made on more complex compositions.

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 on 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. Very little data is currently available on the mechanical properties of MOX. The experimental measurement of the plastic and elastic behaviour of fuels at high temperature and under irradiation (ideally as a function of O/M ratio, microstructure, chemistry and burn-up), as well a deeper understanding of the basic mechanisms involved in the evolution of mechanical properties, are needed for the development of better models to predict mechanical evolution and integrity.

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 "Joint Oxyde-Gaine (JOG)", i.e. the oxide-clad joint, 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, commonly called the "Réaction-Oxyde Gaine (ROG)". 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 five issues need to be known with accuracy for the improvement of fuel performance codes, which play an essential role in the improvement of safety margins and performance, as well as in the qualification of innovative nuclear fuels. These improvements and qualification rely traditionally on integral irradiation testing (full length pins and assemblies) representative of the conditions of GEN IV reactors either in material testing reactors or in the limited number of fast neutron reactors available

and MOX, but mainly for GenII/III applications, for instance Chromox: https://tel.archives-ouvertes.fr/tel-01037885/document

[110] J. Rest et al., J. Nucl. Mater., in press 52018), https://doi.org/10.1016/j.jnucmat.2018.08.019

[111] M. Tonks et al., J. Nucl. Mater. 504, 300 (2018).

[112] L. Luzzi et al., Nucl. Eng. Des. 330, 265 (2018).

today. It then involves the examination of the irradiated fuels, as well as corresponding measurements on fresh fuels for reference. 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 hot-labs 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, 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 $(U,Pu)O_2$ fuel made by a group of European experts [113].

The FP7 ESNII+ project[114] (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. [115] 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 7.

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 with those of the pre-normative research to unveil the missing relevant data and elementary mechanisms underpinning the fuel behaviour, and also to extend the reliability regime of traditionally deduced empirical laws governing various aspects of nuclear fuel under irradiation, which are implemented in fuel performance codes.

This advanced modelling and characterization approach has started later on fuel compounds than for structural materials, the first atomic scale computational studies date to the late 1990s. 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.

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

[114] http://www.snetp.eu/esnii/

[115] K. Tucek et al., New catalogue on $(U, Pu)O_2$ properties for fast reactors and first measurements on irradiated and non-irradiated fuels within the ESNII+ project, FR17, International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development, 26–29 June 2017, Yekaterinburg, Russian Federation.

TABLE 7: *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 $(U,Pu,Am)O_2$ $(U,Am)O_2$ Fresh oxide fuels as reference	Determination as a function of O/M, Pu, Am content, BU of • Melting points • Heat capacity • Vapour pressure Techniques: laser heating, calorimetry, Knudsen cell mass spectrometry (KEMS)
	Evolution of thermal properties		Measurement as a function of temperature, composition (O/M, Pu/M, BU) of • Thermal conductivity • Thermal diffusivity • Emissivity Techniques: laser flash analysis
Atom transport and microstructural evolution	Atom transport		Measurement as a function of temperature and initial composition (O/M, Pu/M, BU) of • Lattice parameter • Diffusion of U, Pu, Am • Diffusion of O • Oxygen potential Techniques: SEM-WDX, EPMA, SIMS, coulometric titration, XRD
	Microstructural Evolution		Measurement as a function of temperature (radial position), initial Pu/M, O/M ratios and porosity and BU of • 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 Techniques: Optical microscopy, SEM/WDX-EDX, TEM, EPMA, SIMS, XRD
Fission Products and Helium	Gas (fission gas and He) behaviour		Measure of gas release in irradiated fuel as a function of BU and irradiation history (normal, off-normal and severe accident conditions) Techniques: gas puncturing, KEMS, thermal treatment, thermal desorption spectroscopy
	Non-gaseous FP transport		Measure of fission products transport, release and compounds formed as a function of BU and history (in normal, off-normal and SA conditions) Techniques: Gamma scanning of fuel pins, KEMS, optical microscopy, SEM/WDX-EDX, EPMA, TEM, XRD, ICP-MS.

Main issue	Breakdown in sub-issues	Materials concerned	Techniques/ Methods
Mechanical properties and mechanical interactions between fuel and cladding	Evolution of mechanical properties and “state” of fuel pellets Fragmentation, cracking	Irradiated oxide fuels and fuel pins: MOX (U,Pu) O2 UO2 (U,Pu,Am)O2 (U,Am)O2 Fresh oxide fuels as reference	Measurement as a function of temperature, Pu/M and O/M ratio, microstructure and BU of: • Thermal expansion • Elastic constants • Ultimate stress • Yield stress • Brittle to ductile transition temperature • Hardness • Creep rates Techniques: XRD, dilatometry, acoustic methods, micro-nano indentation, eventually miniaturised mechanical testing methods
	Fuel Cladding Mechanical Interactions		Measurement as a function of temperature, Pu/M and O/M ratio, microstructure and BU of: • Pellet geometry • Inner pin geometry • Swelling • Pellet density Determination of remaining gap width between pellet and cladding Techniques: Optical inspection, XRD, dilatometry, pin profilometry, immersion method, optical microscopy, SEM, neutron radiography
Chemical interactions between fuel, cladding and coolant	Chemical interactions between fuel and cladding, internal cladding corrosion	Irradiated and fresh fuel pins, Breached pins, Sodium, lead and lead-bismuth coolants	Determination of • Width of corrosion layer • Composition of corrosion layer Techniques: Gamma scanning, SEM/WDX-EDX, optical microscopy
	Chemical interactions between fuel, cladding and coolant in case of severe accident		Measurement (as a function of BU if possible) of: • Size of breach • Amount of fuel / fissile matter loss • Phases produced Techniques: Optical inspection, Gamma scanning, SEM/WDX-EDX, optical microscopy

This type of stud studies started to develop significantly in the year 2000s on uranium fuels, especially UO_2 [110, 116], and to a lesser extent uranium carbide and uranium nitride. They progressed significantly in the F-BRIDGE FP7 project (2008-2012).[117] Basic data such as diffusion coefficients of oxygen, cation and He, melting temperatures and consistent

[116] R. Konings et al., Behaviour and Properties of Nuclear Fuels, in Experimental and Theoretical Approaches to Actinide Chemistry, First Edition, Ed: J.K. Gibson, W.A. de Jong, John Wiley & Sons Ltd ‘2018).

[117] https://cordis.europa.eu/publication/rcn/16699_en.html

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. [118] 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. In particular, recently obtained material properties for oxide, nitride and carbide fuels were implemented in the TRANSURANUS code.[119] Two models for Poisson's ratio and emissivity of UO_2, a specific porosity correction model, as well as a formulation for intra-granular diffusion of fission gas,[120] were also added to TRANSURANUS. The new version of the code was successfully compared against experimental results from the SUPERFACT experiment in the Phénix reactor.

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 amongst others 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)[121] 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 2015[122]. 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, as well as to fuels containing non-gaseous fission products. 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 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 8 (next page).

[118] M. Bertolus et al., J. Nucl. Mater. 462, 475 (2015).

[119] E.A. Kotomin, J. Nucl. Mater. 393, 292 (2009).

[120] P. Van Uffelen et al., Nucl. Eng. Design, 43, 477 (2011).

[121] https://www.oecd-nea.org/science/wpmm/expert_groups/m2f.html

[122] https://www.oecd-nea.org/science/docs/2015/nsc-r2015-5.pdf

TABLE 8: *Main issues concerning modelling and characterization*

Main issue	Breakdown in sub-issues	Materials concerned	Experimental techniques/ Modelling methods
Temperature: Margin to fuel melting	Phase diagrams: phases as a function of composition	Fresh, self-, ion or neutron irradiated MOX $(U,Pu)O_2$ 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)	• 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		• 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		• 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		• 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		• 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		• Modelling of grain growth, fragmentation • Mesoscale modelling of pore evolution, central void formation • Diffusion experiments with high temperature gradient (laser heating)

Main issue	Breakdown in sub-issues	Materials concerned	Experimental techniques/ Modelling methods
Fission Product and Helium behaviour	Gas behaviour	Fresh, self-, ion or neutron irradiated 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)	• 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 neutron irradiation, TDS, SIMS, mass spectrometer coupled to Knudsen cell) • 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		• 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		• 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		• 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		• 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		• Atomistic modelling of thermal expansion in presence of defects and fission products • Measurement of thermal expansion of infused or self-irradiated fuels

Main issue	Breakdown in sub-issues	Materials concerned	Experimental techniques/ Modelling methods
Compatibility between fuel, cladding and coolant	Cladding corrosion: ROG	Fresh, self-, ion or neutron irradiated MOX $(U,Pu)O_2$ 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)	• 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		• 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		• 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 9).

For each case dedicated irradiation testing will also be necessary in material testing or power 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 ^{241}Am pollution due to ^{241}Pu 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 that is more readily dissolvable to close the nuclear fuel cycle.

TABLE 9: *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	• 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)	• 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	• 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		• 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	• 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

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 that is 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.[123, 124]

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.[125, 126]

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 ^{241}Pu 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.[127] 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

[123] R. Parrish et al., J. Nucl. Mater., 510, 644 (2018).

[124] E. Douglas et al. J. Nucl. Mater., 393, 1 (2009).

[125] V. Tyrpekl et al., Rev. Sci. Inst., 86, 023904 (2015).

[126] M.A. Pouchon et al., ETH Zurich, https://doi.org/10.3929/ethz-a-010699574 (2016).

[127] F. Jorion et al., Research and development for the fabrication of minor actinide-bearing fuel materials and technologies, IAEA (2015).

intimate contact with carbon needs to be improved or replaced.[128] This process is relatively simple, but the high temperatures result in powders with very low specific surface area, rendering them difficult to sinter. Comminution 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.[129, 130]

In the case of nitrides, enrichment in ^{15}N during fabrication and its recovery from CO in the off gases needs to be dealt with.[131] Due caution must be given to maintain 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.

Moreover, 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.

[128] Hj. Matzke, Science of advanced LMFBR fuels, North-Holland (1986).

[129] P. Malkki et al., J. Nucl. Mater. 452, 548 (2014).

[130] D. Salvato et al., Ceram. Int. 43, 866 (2017).

[131] N. Chauvin et al., State-of-the-art Report on Innovative Fuels for Advanced Nuclear Systems. NEA - 6895 (2014).

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

In addition to the conventional pellet-in-pin fuel designs of the ESNII reactors, fuels for the Molten Salt Reactor (MSR) designs deserve some attention. 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.[132] The liquid salts must have favourable physical and chemical properties. Then, one of the challenges for the development of this fuel is the optimization of the composition in relation to neutronic and clean-up conditions. Due to the liquid state, traditional fuel synthesis (pellets, pins, elements) is not applicable here. The fabrication aspects will be dominated by the synthesis via fluorination of the fuel components, and their purification from impurities such as oxygen and water. The scientific and technical issues relevant to the in-reactor behaviour of this type of nuclear fuel under irradiation are very different from solid fuels. Radiation effects are expected to play little role in the fuel behaviour, 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. Many fission products dissolve in the liquid fuel, and are strongly bonded. Noble gases and the noble metals are exceptions, and their removal during operation is an important issue. Experimental demonstration of these characteristics is important and requires the development of irradiation facilities to study the fuel behaviour during irradiation in static and dynamic conditions (loops). 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 involved in the development of advanced fuel materials are listed in Table 9.

[132] J. Serp, Prog. Nucl. Energy 77, 308 (2014).

4

CROSS-CUTTING ISSUES

4. CROSS-CUTTING ISSUES

4.1. Objectives

The qualification of materials in environments corresponding to the expected service conditions, the development of models capable of anticipating materials degradation in operation, as well as the development of new material solutions for better performance and fitness for purpose, 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 maintenance costs. The EERA JPNM firmly believes that **cross-fertilisation on materials between different technologies can only be beneficial and contributes to accelerating progress**. Moreover, joint research on materials topics of common interest, sharing methodologies and facilities, is an effective way to make better use of available human, infrastructural and financial resources.

Many common or contiguous research topics exist with other nuclear fission technologies, namely GenII/III reactors and (very) high temperature reactors for co-generation, but also with fusion energy technology. The EERA JPNM is in the privileged position of acting as catalyser for collaboration between these two nuclear technology branches. Thus, significant effort was deployed to dialogue with the platforms involved, NUGENIA and NC2I (exploiting the MoU signed with SNETP) and also EUROfusion,[133] to identify cross-cutting issues.

Commonalities exist with energy technologies outside the nuclear field, as well. While each energy system faces different materials' challenges, cross-cutting issues with nuclear materials can be identified, in particular concerning materials that operate under extreme conditions (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.

[133] The H2020 project M4F is a first example of a project jointly run by fission and fusion materials communities

4.2. Commonalities with nuclear GenII/III reactor materials

The first common area between GenII/III and GenIV nuclear technologies concerns integrity (performance and ageing) of structural materials. Four topics for collaboration have been identified (for more details see Technical Annex TA.2.1[71]):

- 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.

A second area of close collaboration between GenII/II and IV communities is the research on nuclear fuels and claddings (see Technical Annex TA.2.1[71]). 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
- Innovative fuels and synthesis routes
- Fuel performance codes development and validation

Other commonalities are:

- Modelling, including separate effect experiments and materials characterization
- Materials and coatings considered for accident tolerant claddings
- Use of experimental facilities enabling the manufacturing and characterisation of fuels

Finally, three potentially common research areas are found also with the field of design of innovative LWR (for more details see Technical Annex TA.2.3[71]):

- Supercritical Water Cooled Reactor (SCWR) materials
- Development and application of advanced or novel materials manufacturing processes
- Development of new materials more resistant to corrosion

Table 10 summarizes materials issues of potential cross cutting interest between EERA JPNM and GenII/III (NUGENIA), as well as NC2I (see next chapter).

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 are subjected to the same requirements, namely high temperature operation in contact with flowing pressurized helium. Therefore the materials of interest are largely the same (for more details see Technical Annex TA.3[71]):

- High temperature resistant materials for intermediate heat exchangers, insulating structures, control rods, and other internal structures
- Design life of 60 years with the development of relevant design rules and design curves and codification
- Study of the degradation of the properties of relevant materials in operation, due to the synergistic effect of high temperatures, mechanical stresses, radiation and gas coolant environments

Table 10 summarizes also materials issues of potential cross cutting interest between EERA JPNM and VHTR (NC2I) communities.

TABLE 10: *Summary of potential cross-cutting issues on materials through nuclear fission technologies*

General topic	Commonalities
Integrity of structural materials	• 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 • Charged particle 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	• 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	• Materials issues for SCWR • Development and application of advanced or novel materials manufacturing processes • Development of new radiation-, corrosion- and high temperature resistant materials
(V)HTR materials	• Issues related with development and qualification of high temperature resistant materials in gas coolant environment

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, that concern **the only possible cross-cutting class of materials between fission and fusion, namely F/M steels** (for more details see the Technical Annex TA.4[71]):

- Compatibility with heavy liquid metals: corrosion and liquid metal embrittlement and a need for protective (alumina) surface layers
- Codification of F/M steels, as these are not yet included in design codes such as RCC-MRx, except at probatory level (specific issues are cyclic softening, plastic flow localisation, ...); joint materials database
- Welding procedures and characterization to detect defects and evaluate their consequences, taking into proper account the presence of residual stresses
- Small specimen testing in connection with small volumes of materials that can be exposed to irradiation
- Development of F/M steels with better high and low temperature properties, with the 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
- Establishment of a methodology to screen among radiation-resistant materials, e.g. by combination of charged particle irradiation, coupled with suitable PIE and, in particular, mechanical property probing techniques on small volumes
- Advanced modelling to continually improve the level of understanding of materials behaviour by exploiting advanced microstructural characterization and evolving physical models

Table 11 summarizes materials issues of potentially cross-cutting interest between GenIV fission and fusion.

4.5. Commonalities with materials for other energy technologies[134]

The commonalities with other energy technologies can be classified in two groups, one generic and one specific, namely:

1. General **methodological common patterns**
2. Materials for **high temperature** applications

Concerning the first group, an analysis of materials needs and materials science approaches through energy technologies in EERA, including nuclear, led to the identification of the following **common patterns**[135] (for more details, see Technical Annex TA.5.1[71]):

- Ageing and degradation mechanisms studied by combining advanced experimental characterization with multiscale modelling
- Characterization of energy materials and devices: contribution of large scale facilities, as well as in situ and operando techniques
- Rational design of materials supported by modelling

The second group stems from the recognition of the fact that **resistance to high temperature is a requirement for materials in a wide spectrum of energy technologies**, because the efficiency of thermodynamic cycles operating between two heat reservoirs is improved by increasing the temperature of the hot one and some energy production systems inherently require high temperature 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 also need to operate at the highest temperature possible to increase efficiency and minimize emissions.

At high temperature, environmental aggression seriously limits the performance of materials and the lifetime of components, so the two issues of high temperature operation and compatibility with aggressive environments cannot really be separated. The components and the classes of materials of interest, as well as the properties that need to be evaluated and improved, are largely the same through the different energy technologies, thus there are serious grounds for the establishment of a joint materials qualification and development research pool, that would share facilities and resources, exploiting also modern modelling techniques.[136] More details on the commonalities concerning high temperature materials are given in the Technical Annex TA.5.2.[71]

Within high temperature applications, **specific commonalities identified and targeted between materials issues for GenIV fission energy and concentrated solar power (CSP)**. Details on this point are given in the Technical Annex TA.5.3.[71] Table 12 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 and environmental degradation resistant materials.

[134] This section 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/filesharer/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/filesharer/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.

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

[136] https://www.eera-set.eu/wp-content/uploads/FINAL-June-2018-EERA-EUMAT-position-paper-HT-materials.pdf

TABLE 11: *Summary of GenIV fission/fusion potential cross-cutting issues on materials*

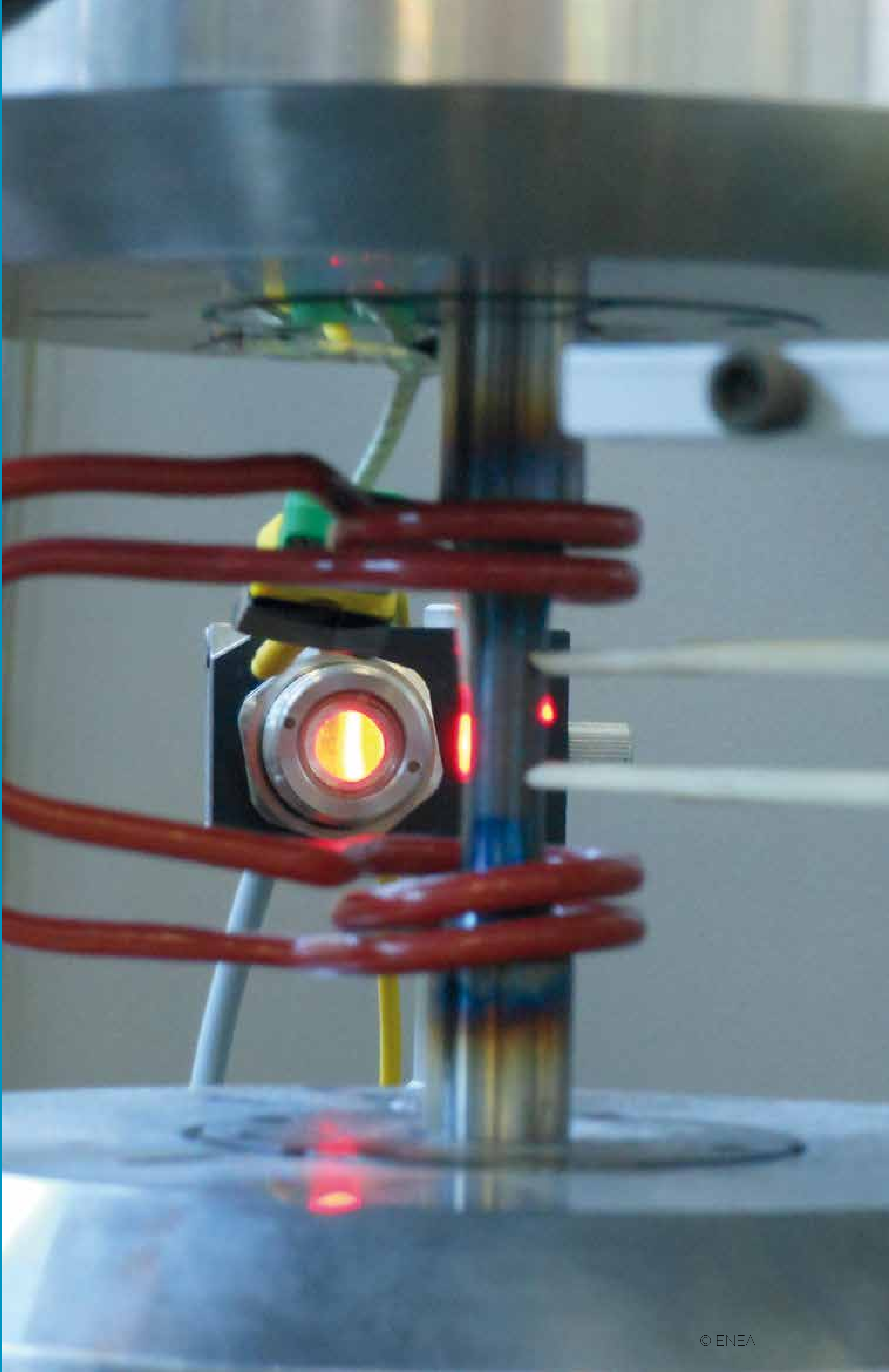
General topic	Commonalities
F/M steels for current concept designs	F/M steels • Design rules in RCC-MRx for F/M steels • F/M property assessment at low and high temperature: plastic flow localisation, cyclic softening, thermal creep, thermal fatigue, creep-fatigue, • F/M welding • F/M compatibility with HLM: corrosion, erosion, LME, and test standardization
Innovative high temperature resistant steels	• Optimisation of F/M ODS & TMT steel fabrication ◦ 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	• Ceramic coatings on steels (alumina or aluminium containing)
Physical modelling and modelling-oriented experiments	F/M steels • 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, …

TABLE 12: *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	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,…)	• Corrosion/oxidation/dissolution/erosion processes • Coatings of proven stability • Self-healing surface protection mechanisms
Other properties to be maintained	• 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	• Creep-resistant and corrosion resistant (FeCrAl) F/M steels (including ODS) • Austenitic steels • Ni-based alloys
Refractory materials: metals and ceramic composites: existing and advanced	• V-, Mo-based alloys • SiC_f/SiC • Alumina based ceramics • Max phases
Materials qualification	• Exposure facilities • Test standardization
Advanced modelling and characterization	• Physical mechanisms

5

WORKFLOW AND TIMELINE



5. WORKFLOW AND TIMELINE

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 and materials solutions, when design solutions are insufficient or unsatisfactory.
- Fundamental research on materials behaviour mechanisms provides tools to better assess component lifetimes and suggests routes towards 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 solutions are explored.

The virtuous circle of Figure 2 represents well this continuous process. In this SRA, also a different representation of the workflow can be provided, in which **two parallel routes supported by modelling are pursued**, one for the **shorter term application (demonstrators)**, the other for the **longer term (FOAK and then commercially deployed GenIV reactors)**. This as illustrated in Figure 8, which shows that:

1. **Demonstrators will be built using existing materials**, that need to be qualified for the expected service conditions and, in turn, determine the allowable service conditions due to their limitations in terms of performance. Thus the demonstrators will not offer the best efficiency and will be unlikely to meet all the requirements of GenIV systems;
2. FOAK prototypes and then **truly GenIV commercially deployed systems will need innovative material solutions** in order to achieve the expected targets of efficiency, optimal use of resources, waste reduction, and economy or, in short, sustainability.

Figure 8 summarises the crucial issues for each ESNII system, concerning both structural and fuel materials, clarifies that physical modelling is the crucial support for both paths and emphasises the overarching principle of safety.

Figure 8 also highlights that, depending on the system, some materials and issues are more relevant than others. It is therefore possible to associate indicative milestones for the qualification/development of specific material solutions with the moment when it needs to be ready for application, in order to enable

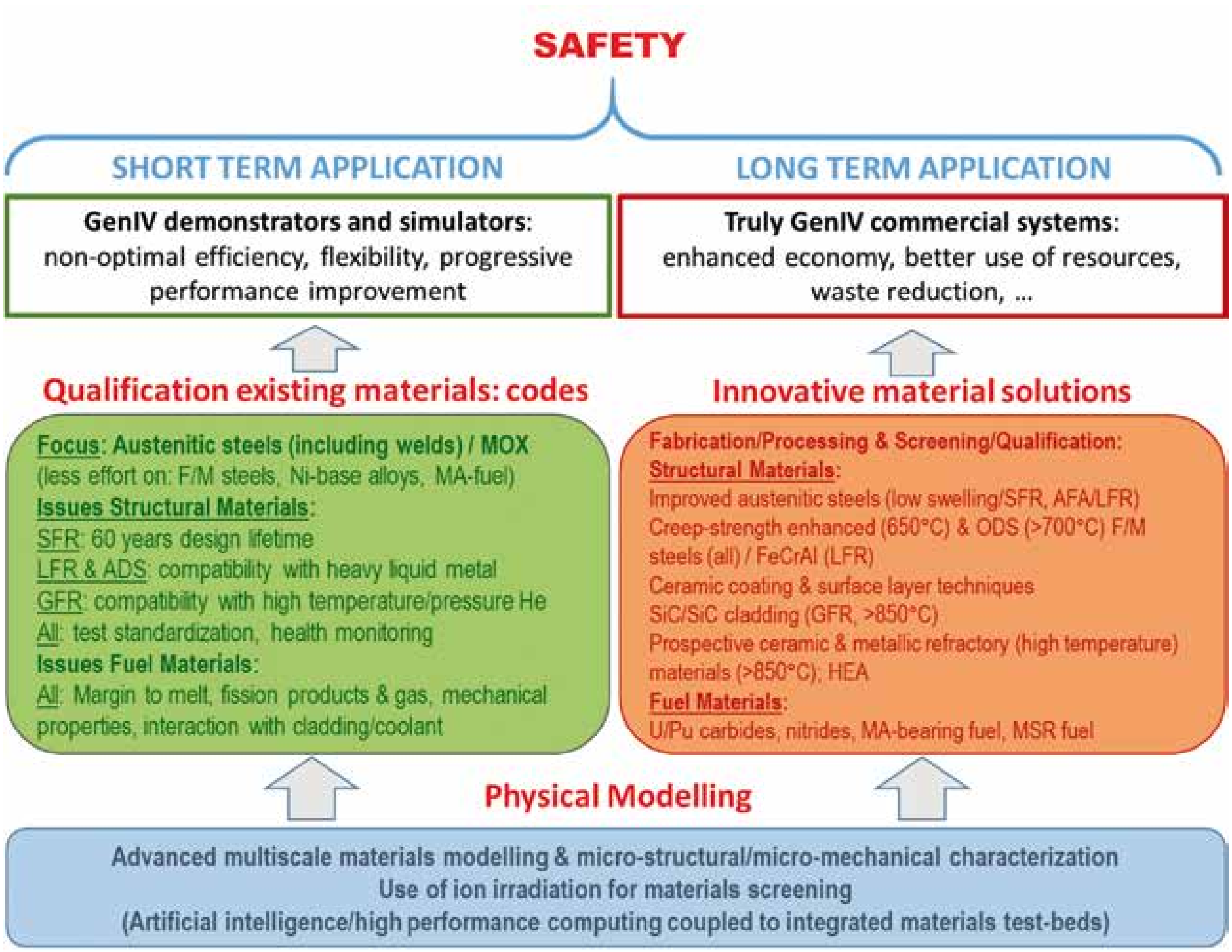


FIGURE 8: *The two parallel paths followed in the present SRA, one with short term and another with long term perspective.*

the design of a specific system. However, it is impossible to establish precisely the timeline according to which the different possible systems will be designed, licensed and built, because this will depend on political and technical decisions that are out of the scope of the present SRA.

Figures 9 and 10 provide a **simplified timeline**, for structural and fuel materials respectively, defined based on an arbitrary "chronological" order, where the milestones are given by the fast reactor system development,[137] associated with the different TRL attributed to each of them,[12] rather than on an actual timeline expressed in years. Each row refers to experimental work and supporting models aimed at a specific goal, as expressed. The longest arrows are those projected farther in the future and are more uncertain, but are expected to benefit from the operation of the demonstrators. If the goal has to be reached on time, all activities should start now (in most cases they 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 milestone. These charts are of course only indicative and will require progressive updating over time.

[137] In particular, at the time of writing of this SRA the fate of ASTRID is undefined.

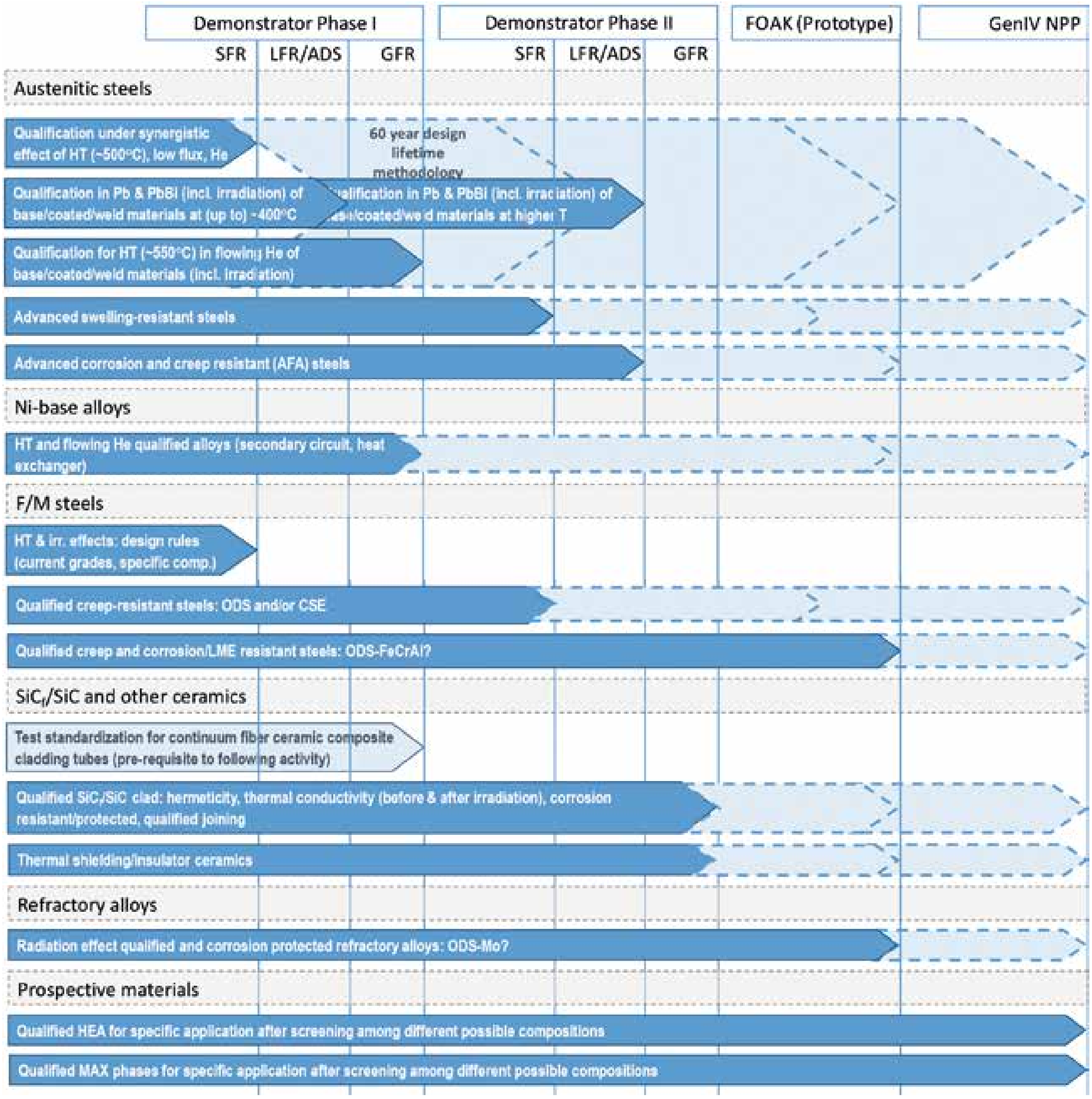


FIGURE 9: *Schematic timeline for structural materials. Dashed outline arrows indicate potential continuation of activities towards following milestones.*

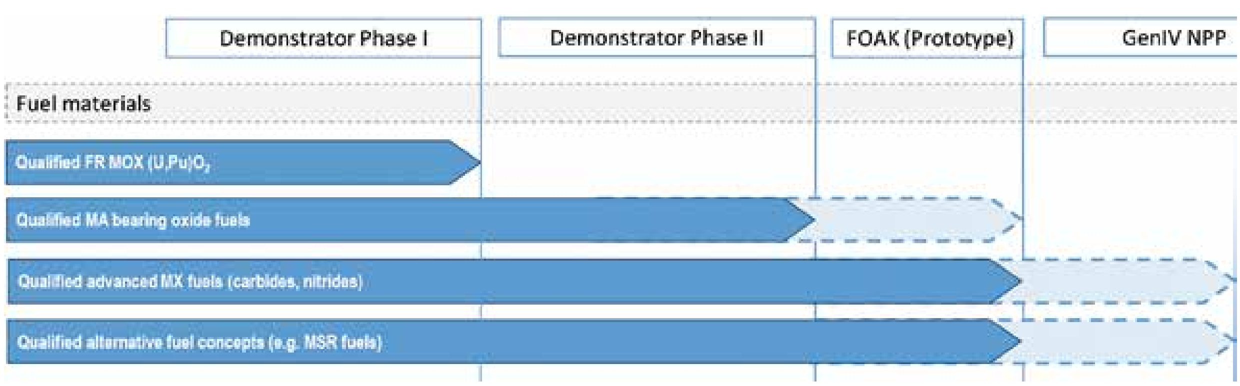


FIGURE 10: *Schematic timeline for fuel materials. Dashed outline arrows indicate potential continuation of activities towards following milestones.*

6

INFRASTRUC-TURES FOR NUCLEAR MATERIALS R&D

6. INFRASTRUCTURES FOR NUCLEAR MATERIALS R&D

The research activities described in sections 3 and 4 have as an essential prerequisite the availability of suitable facilities and infrastructures for materials qualification through exposure to conditions representative of service 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**. The first two can be of interest and application also for non-nuclear technologies (e.g. facilities for tensile and fracture toughness tests, standard creep and corrosion tests, slow strain rate tests in environment…) and may be in some cases even redundant. Moreover, microstructural studies on materials exposed to corrosive environment and/or high temperature can be performed in many laboratories. In contrast, **irradiation is quite specific of nuclear materials** and **irradiation facilities are scarce**, as well as are scarce the infrastructures that allow **testing and examination of irradiated materials**. Therefore, this section focuses mainly on facilities for exposure to irradiation and handling of irradiated materials.

Notwithstanding the focus of this section, it is clear that equipment in **furnaces** to perform for example very long-term creep and creep-fatigue testing, or **loops** for exposure to flowing coolants, are **crucial facilities** that are in fact not so common and may not be available in sufficient quantity for a full qualification of materials for sustainable nuclear energy. This section summarises the main features and issues related with the facilities for irradiation and handling of irradiated materials. More details can be found in the Technical Annex, section TA.6,[71] which is an extended version of the present one.

6.1. *Irradiation Facilities*

6.1.1. Fast Neutron Facilities

Neutron irradiation facilities and associated 'hot' cells laboratories are a central necessity for performance and safety testing of all types of structural and fuel materials. Since the ESNII demonstrators are all fast neutron spectrum systems, materials for their construction should be qualified in fast neutron irradiation spectra. Unfortunately, there is currently no fast neutron power or testing reactor operating in Europe. 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 demonstrators within the coming two decades is thus essential in order to provide Europe with a facility to qualify structural and fuel materials for commercial GenIV reactors.

6.1.2. Material Testing Reactors and associated 'hot' cells

Materials testing reactors (MTR) are high power research facilities that can be used to expose materials to operation-like conditions. Flexibility and ability to adapt to changing needs is a fundamental principle for such reactors. MTR can perform a variety of test irradiations simultaneously, the number of these depending on their design and on the nature of the tests. In specific cases, MTR 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 MTR 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 MTR currently operational in the world, Europe hosts seven (see Table 13). Two of these can be used for fuel testing and have appropriate 'hot' cells available. The others do not have currently a license to handle fuel, but are available for structural materials' investigations. These centres are equipped with 'hot' cells, as well. **Europe's neutron irradiation capacity is currently very limited**: **the opportunities to test new materials in reactors are therefore restricted to the extreme**.

TABLE 13: *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 *)	Dose rate [dpa_{Fe}/fpy]	Accessible temperatures [°C]
BR2	Mol, Belgium	7	Up to 5	50-1200
HFR	Petten, The Netherlands	5.1	<7	80-1100
LVR-15	Řež, Czech Republic	3	~1	50-850
MARIA	Świerk-Otwock, Poland	1	~1	50-100
Triga Pitesti	Pitesti, Romania	1.8	2.5	80-300 (→ 500 future)
Triga Mark II	Ljubljana, Slovenia	0.06	< 0.01	20-50 (→ 300 future)

(*) https://nucleus.iaea.org/RRDB/RR/ReactorSearch.aspx

For irreplaceable component materials, the doses expected at the end of life can be reached in existing MTR, 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, MTR 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, as described in the next section.

6.1.3. Charged particle irradiation facilities

Charged particles can be electrons, protons or ions of different weight. Although the penetration of protons and light ions can be significant, all these particles are obviously efficiently stopped by electrons and ions inside the materials, thereby affecting only limited volumes, insufficient to produce standard specimens for e.g. mechanical property testing. Moreover, there are spectral differences with respect to neutrons in terms of damage that is produced and the progressive slowing down results in damage production gradients, while chemical species initially absent in the material and/or atoms in excess are injected in the target material, including sometimes unwanted impurities that are difficult to control, e.g. carbon. Despite these shortcomings and limitations, which certainly prevent their use for full qualification purposes, charged particle irradiation is useful to get 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 modelling and screening purposes, exist in Europe at nuclear research centres and universities, including one facility usable for fuel.

6.2. Handling of irradiated materials

6.2.1. 'Hot' Cells and shielded facilities[138]

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. 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 often 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). In addition **some centres are equipped with 'hot' cells even without possessing an MTR**. Yet the number of centres that dispose of facilities of this type remains limited and in several cases the maintenance cost are expected to lead to closure without replacement. In particular, the availability of 'hot' cells suitable for fuel handling is very limited and endangered. On the other hand, **the number of sites where new 'hot' cell facilities have been recently built or refurbished is limited**.

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 (see section 6.2.2) 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

[138] In addition to those associated with MTRs.

fuel procurement, since these facilities are indispensable for the preparation and safety testing of new fuel.

In conclusion, 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.**

6.2.2. 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 currently 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.

6.3. 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 competiveness 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. 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**. Currently, while fusion research can rely, as part of the EUROfusion consortium, on a funded HCP facility, **no equivalent exists for fission research**. This is a foremost condition for advanced modelling to bring the appropriate support to the development and qualification of the materials needed for, especially, GenIV systems.

6.4. General considerations on nuclear infrastructures in Europe

6.4.1. Costs related to crucial infrastructures build and use

The construction and maintenance of infrastructures and facilities such as MTRs and 'hot' cells is beyond 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,[139] 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€/yr.

6.4.2. Renewal of nuclear infrastructures

A research agenda can only be fulfilled if it is matched by appropriate and timely available infrastructures. Therefore, in the present context, **in order to maintain its competences and progress further in the field of nuclear materials, 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 demonstrator);
- 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;
- Ensure appropriate access to **high performance computing for the fission community**;

In addition, three recommendations for policy-makers, that are further developed in the next section, are in order:

1. **Plan** major infrastructures judiciously at a **pan-European** level in a **harmonised** way;
2. Implement consequent **open access** and infrastructure **sharing** initiatives;
3. Foster **joint programming** (fusion energy provides an example to follow).

6.4.3. Sharing and joint programming of nuclear infrastructures

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

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

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, as well as 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 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).

To facilitate both sharing and mobility, in a context of scarcity of infrastructures and financial means, and in order to foster pan-European harmonised infrastructure planning , **joint programming is key**. In particular, it is believed that the creation of a **permanent joint European planning forum and management scheme of nuclear infrastructures** is a timely need.[140]

The **joint forum** should mainly agree upon **irradiation future needs**, in order to plan and propose **harmonised European investments**. It may also express views on the rules for financial coverage for the use of nuclear infrastructures.

The **joint scheme for the management of nuclear infrastructures** should fulfil the following conditions:

1. Be a **single entry point**, i.e. a single interlocutor whom users should address for the use of available infrastructures;

2. **Coordinate the best use of infrastructures** for a given experiment, by pooling facilities and experts in Europe, to design the best experiments possible (also the cheapest ...), establishing if and when there are advantages to access non-EU infrastructures;

3. Benefit from the **harmonised commitment of MS to pay the costs of the infrastructures and use**, as part of the engagement to innovation, but develop flexible management attracting collaborative projects and also industrial users;

4 **Distinguish between R&D users** (low cost open access, selection based on merit of proposal, open access data and results, ... also open to non-MS) **and industrial users** (charged services, data protection, ...).

Within such a scheme, tenders could be launched and/or differences between available facilities could be optimally used to explore specific effects, including, whenever suitable, fundamental studies and the use of ion irradiation. In particular, such a scheme would allow the **best use of available space in reactors** to be made, through the design of joint campaigns. For instance, while it is true that MTR operation is expensive and cannot be offered for free, irrespective of the open access nature of the facility, under some conditions it is possible to perform inexpensive "piggy-back" irradiation experiments, in a concerted framework, especially if experiments funded by industrial users can be partially used for this purpose at no detriment of the customer.

Of course, for this scheme to see the light the **willingness of sharing and joint programming must exist at all levels**, including high management of research centres as well as MS policy-makers. This should also be properly fostered by Europe-driven actions. It remains to be seen whether the conditions for this approach to materialise actually exist.

[140] This could be a European version of the American NSUF: https://nsuf.inl.gov/

7

EDUCATION, TRAINING AND MOBILITY OF RESEARCHERS

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 in contrast 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, **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 nuclear energy projects, integrating expertise from other fields into specific nuclear applications**. GenIV plays here an important role, by involving 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 nuclear competences. In this respect, **infrastructure sharing is a key driver, to be made effective within joint programming schemes** (see section 6.4.3). 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 (ECTS)**. To be attractive, **it should enable the development of knowledge and skills in 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**.

In order to reach this goal, the EERA JPNM considers that a **coordination in terms of 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 14.

TABLE 14: *Description of a possible E&T programme on nuclear structural and fuel materials, as envisioned by the EERA JPNM.*

What	• A learning package - E&T program devoted solely to nuclear materials (structural materials and fuels)
When	• Annually or bi-annually, at predetermined dates
Participants	• Master's students from relevant engineering disciplines • PhD students • Employees from the nuclear industry
Class size	• Class size is optimally 20 people, max. 30 people
Where	• 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	• 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	• 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 • E-learning whenever suitable
Topics	• 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	• 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	• A short course to motivate applying successful teaching methods • Guidelines for desirable lecture composition and assignment design
Suggested curriculum features	• 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	• 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

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 in which, ultimately, materials are going to be actually used and therefore fully qualified, based on return of experience, in the proper operational environment. It is however **essential to have a close link to the industrial application**, because in terms of technology readiness level (TRL[6]) 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.

Three classes of stake-holders are the **industrial counterpart of the EERA JPNM**:

A. Reactor designers and constructors: The **EERA JPNM has to know very clearly the goals and the needs of the reactor designers and constructors**, in order to orientate the research towards the support of the processes of licensing and construction of advanced nuclear systems, for which return of experience is limited or even non-existent. In addition, **the results of the research of the EERA JPNM need to be appropriately and efficiently transferred to the reactor designers and constructors**, in the form of data/knowledge, through suitably updated and managed databases (see section 2.4). Fast reactor designers and constructors are almost all involved in ESNII, which explains the strong link with ESNII.

B. Reactor operators (nuclear electro-producers): The reactor operators are the customers of the reactor designers and constructors, they essentially decide whether or not innovative nuclear systems, and which ones, are attractive for their business. In particular, they decide about the economic relevance of closing the fuel cycle,[141] although political decisions have also an influence there. Finally, reactor operators are those that may provide return of experience on materials in service

[141] Nuclear electro-producers have currently drawn back from investing in technologies to close the fuel cycle because of uncertain future of nuclear energy in some countries and in Europe at large and because of renewable penetration leading to load follow mode. Incomes are reduced, while cheap fossil fuel energy sources remain convenient (especially coal). **Therefore without public and political support, also financial, towards sustainability of nuclear, GenIV technologies will have difficulties to emerge on the nuclear arena**. Nonetheless, even if industrial interest is shrinking at the moment, the research on materials at TRL<5 remains crucial to enable the development of simulation codes, design codes etc.

behaviour (whether relevant or not for innovative systems). Although designers and constructors are those mostly interested in interacting with reactor operators, as a research platform **the EERA JPNM has to be aware of the industrial experience and needs concerning materials in service, as well as to know towards which innovative solutions they consider moving**. Operators are mainly involved in NUGENIA, hence the importance of establishing a close relationship with this platform.

C. Materials manufacturers: **the process of development of new materials, especially when innovative fabrication routes are explored, eventually requires upscaling 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. Moreover, innovative fabrication processes (for instance additive manufacturing) are currently emerging, the industrial use of which should be enabled. Thus the EERA JPNM should strive to **work hand-in-hand with the materials manufacturing industry, as early as possible along the materials development route**, in order to take into account industrial production upscaling as a criterion. Several platforms include materials manufacurers, for example ESTEP (steel-makers)[142], EMIRI[143], EUMAT[144], …

In addition to industries, **safety authorities and regulators, or at least technical and scientific support organizations (TSO), would ideally be an important counterpart for the EERA JPNM.** Although the interaction with regulators and TSOs is mainly the duty of the system designers, because of the need to refer to specific designs and service conditions in connection with the licensing, it is considered **beneficial for regulators and TSOs to follow the procedures used for materials qualification and possibly guide them from a safety point of view, in order to eventually accelerate the licensability of nuclear components**. However, contacts with TSOs and regulators are quite difficult to establish.

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

1. **Industries may join the EERA JPNM as associate members**. Currently a few industries (reactor designers/constructors, operators, and materials manufacturers/steel-makers), participate directly 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 intellectual property right –IPR- rules and based on the signature of a non-disclosure agreement) **are invited to join the stake-holders' group (SHG) of projects under the umbrella of the EERA JPNM**.

Stake-holders are expected to be involved in task forces aimed at drawing specific research plans.[145] The wish for the future is that also TSOs and regulators may join SHGs of EERA-JPNM projects.

[142] https://www.estep.eu/estep-at-a-glance/

[143] http://emiri.eu/about

[144] http://www.eumat.eu/

[145] See for example the task force report on 60 years operational life of reactors, available at http://www.eera-jpnm.eu/?q=jpnm&sq=nboard

9

INTER-NATIONAL COOPERATION

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 insufficient, or 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:[146]

- **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 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

[146] Clearly, these are first and foremost areas where closer cooperation within Europe would already make a difference. Here they are put in a more general international framework.

above one. Ideally, the database produced in the framework of EERA JPNM could become an official database for the design and construction of GenIV reactors.

- **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, as well. 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 (some of) 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 MTRs, but also other materials exposure facilities: high temperature and coolants), based on existing lists and maps; optimised and complementary use of these facilities, driving similar facilities to non-overlapping uses; design of large joint experimental programmes of 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, e.g. OECD-NEA or IAEA, is pivotal in this respect, to facilitate cooperation on nuclear energy at global level. For this reason, EERA signed a MoU with OECD-NEA, 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.[147]

In concrete terms, **a more effective interaction with the GenIV International Forum (GIF[148]) is advocated**. GIF is a cooperative international endeavour based on a Charter that was signed first in 2001 by a number countries, that has now grown to 14. While some countries are "dormant" members, especially active are United States,

[147] https://www.oecd-nea.org/ndd/ni2050/

[148] https://www.gen-4.org/gif/jcms/c_9260/public

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 a signatory of the GIF Charter, although only some 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.

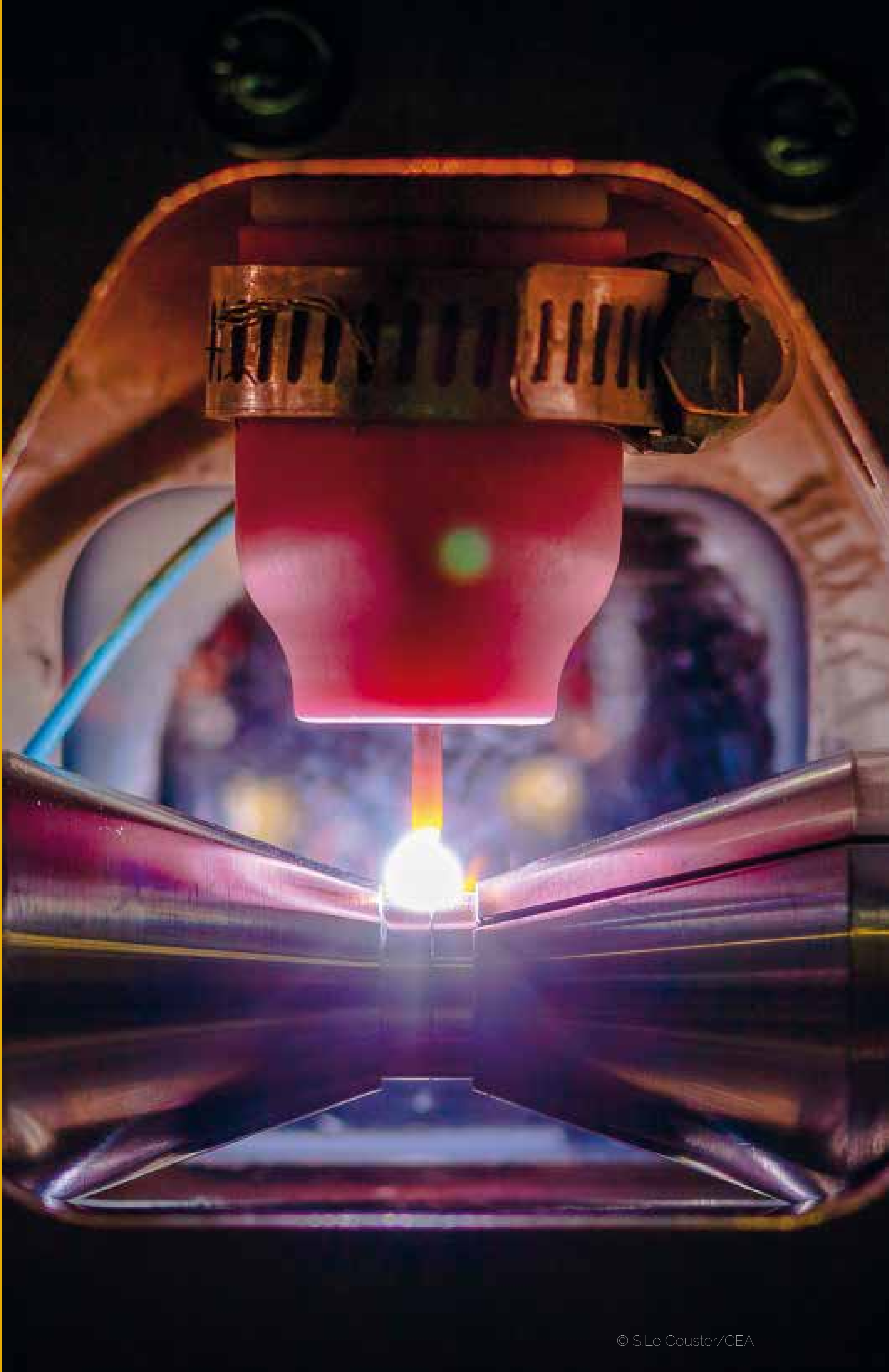
Benefit is quite obviously expected through collaboration with non EU countries involved in the GIF, so USA, South Korea, Australia, Japan, China and Russian Federation, perhaps also Canada, because all these countries have specific activities on GenIV materials. Data sharing is the essence of the GIF, but most countries should also be interested in harmonizing testing and characterization procedures, without which data sharing becomes less meaningful. Collaboration with the US on several fronts, from materials qualification to modelling and development of new materials, is relatively easy and instruments such as I-NERI[149] are in place, although they are not in practice very attractive, especially on the European side.[150] 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 underway 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 all 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 members 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.

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

[150] While in the US this scheme leads to chances for additional funding, this is not the case in Europe. While collaboration with international key partners is always useful, there is often reluctance to disclose costly data, especially without any additional financial incentive.

10

RESOURCES



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 the development and release of such technology, 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 here taken for granted, one can assess that:

In order to adequately address the most important issues, strictly related with the needs of the ESNII demonstrators and their follow up (so, fast GenIV reactors, see Figure 9

Figure 10), 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 cost of about 12-15 M€ per year should be budgeted**.

- 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 (future power reactors, but also other systems such as 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**.[151]

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 is crucial**. MS funding for research activities on nuclear materials for

[151] 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.

innovative systems 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 substantial MS involvement. However, **there is currently no coordinated commitment to earmarked MS funding through Europe**. In contrast, this exists, for instance, in the case of fusion energy, which is taken here as a successful example.

Fusion energy receives support under different forms, including through joint undertakings (Fusion for Energy, F4E) and international agreements (ITER, Broader Approach). Another form of support, in the EU, is through the co-fund instrument called **European Joint Programme** (EJP), namely EUROfusion. This instrument appears to be especially suitable to enable the joint implementation of a roadmap, using, among other tools, internal calls for projects, schemes of E&T&M, etc. with clear MS commitment combined with Euratom support. Of course, such a funding scheme for fusion energy is explained by the widespread support to this research through essentially all EU Members States.

It is here suggested that **nuclear materials at large are a subject that can also find widespread support for an EJP through several MS**, by virtue of its cross-cutting nature (see section 4). The present SRA is a good starting point for this, as it shows the existence of an already established and thriving European nuclear materials research community, partly under the umbrella of the EERA JPNM, and partly also under the umbrella of SNETP pillars, NUGENIA *in primis*. There are also strong connections and overlaps, in practice, between researchers working in nuclear fission and fusion materials. It is therefore believed that **an EJP, or any equivalent co-fund instrument, could be suitable also to support the established nuclear materials community**, with substantial advantages in terms of optimal use of resources, by avoiding duplications and redundancies, as sometimes happens between fusion and fission. However, without support for projects specifically aimed at designing and constructing GenIV demonstrators, such as those considered in ESNII, the driving force to sustain activities on nuclear materials for GenIV will be scarce, beyond fundamental research motivations.

Figure 11 shows the approximate subdivision of resources through EERA JPNM activities, based on 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.

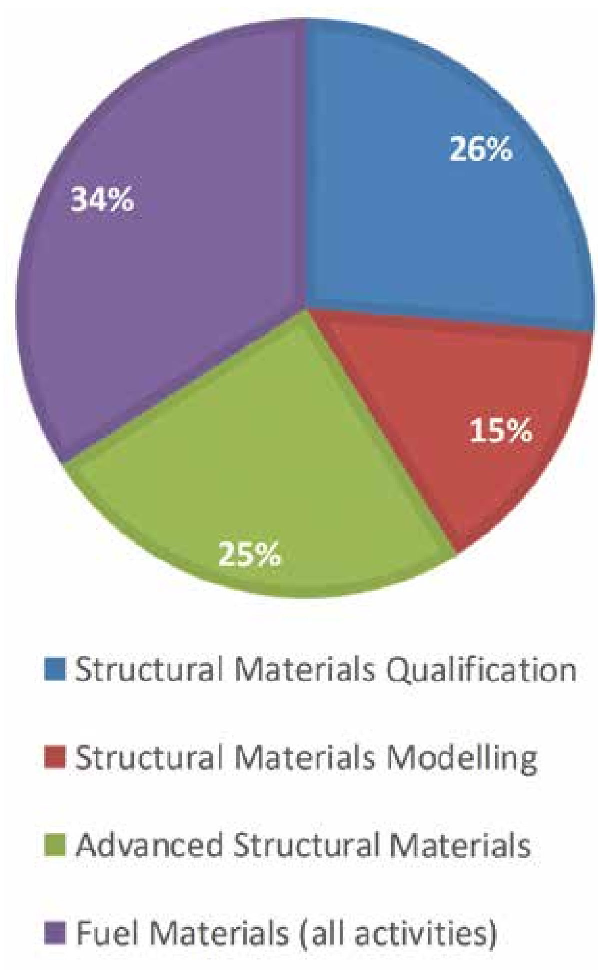


FIGURE 11: *Indicative estimated subdivision of funds between research activities, based on past experience, assuming all activities are sufficiently funded.*

11

RISK ASSESSMENT

11. RISK ASSESSMENT

This SRA suggests a path to be followed to ensure that suitable materials are made available for the design, licensing and construction of GenIV demonstrators, with a view also to the longer term construction of FOAK prototypes and then deployment of commercial reactor. However, all activities described in this SRA are exposed to sometimes very critical risks. Table 15 (next page) provides a list of those that are perceived as more prominent in a number of categories addressed in this document.

TABLE 15: *Risk assessment table on several issues addressed in this document.*

No.	Category	Risk denomination	Risk description	Likelihood(*) and Level of criticality of consequences	Possible consequences	Mitigation strategy
1	GenIV systems in Europe	Low industrial interest to close the fuel cycle	Closing the fuel cycle is the main reason driving the deployment of fast reactors. While this is considered as a necessary step towards nuclear energy sustainability, the investment required, particularly if not supported at political level and/or by public funding, is likely to lead the current nuclear industry to delay the goal of closing the fuel cycle to farther times.	60% Very critical	The development of fast reactors, and GenIV systems in general, may be perceived as a lesser priority for Europe, leading to a significant reduction of financial support.	The activities can shift towards more cross-cutting issues with GenII/III, fusion or even other energy technologies
2	GenIV systems in Europe	ESNII projects stopped	One or more of the ESNII projects might be stopped because of lack of financial or political support.	60% Currently very critical for ASTRID. ALLEGRO might also run into difficulties	Some materials or some specific research lines (e.g. compatibility with a specific coolant) would lose importance and maybe have to be abandoned	Focus on other systems or on cross-cutting issues
3	Prenormative research on structural materials	Insufficient qualification of reference materials and design rules & curves for HLM compatibility	Design Rules and Codes for HLM compatibility require a comprehensive and complete long-term test programme that includes reference materials and welds with relevant engineering and mechanistic models. Data obtained within the available timeframe or with the available resources/infrastructures may not be sufficient for a demonstration of compatibility at design level.	50% Critical	HLM compatibility is critical for licensing of HLM cooled reactors. Without appropriate demonstration at design level, license for construction may not be given. Or the system, due to excessive conservativism, might turn out not to be viable from the point of view of economy/efficiency	Drastic design changes may help to guarantee sufficient conservativism, within available data.

No.	Category	Risk denomination	Risk description	Likelihood(*) and Level of criticality of consequences	Possible consequences	Mitigation strategy
4	Prenormative research on structural materials	Insufficient qualification of reference materials and design rules & curves for 60 years operation life	Long-term tests (> 5 years) for representative service conditions under very stable test conditions need to be performed to qualify materials and components for 60 years design life. These tests require long-term commitments by participating organizations which may dwindle over time. Reliable methods for accelerated tests need to be devised and their feasibility to 5represent long-term degradation needs to be demonstrated by integration of modelling and experiments. The extent to which this is possible is an open issue.	70% Critical in some areas	Lack of design rules and codes for 60 years operation lifetime, with reduced return of investment for nuclear reactors and hence reduced willingness to invest.	Drastic design changes may help to guarantee sufficient conservativism, within available data.
5	Qualification of fuels for GEN IV reactors	Insufficient qualification of MOX fuels for GEN IV reactors	Qualification of new fuels or known fuels in new conditions or with new safety requirements is a long and expensive process which climaxes in an irradiation in reactor. This requires long-term commitments by participating organizations. Also possible that the time for full qualification exceeds the time needs	40% critical	The efficiency of the plant will not be optimal, not fulfilling the GenIV requirements. The system, due to excessive conservativism, might turn out not to be viable from the point of view of economy/efficiency	Better known fuel, such as UO_2, could be used. Qualification done by non-European countries for their own reactors could be used
6	Qualification of fuels for GEN IV reactors	Insufficient qualification of transmutation fuels for GEN IV reactors	Qualification of new fuels is a long and expensive process which climaxes in an irradiation in reactor. This requires long-term commitments by participating organizations. Also possible that the time for full qualification exceeds the time needs.	60% severe	The closed fuel cycle will not be achieved, not fulfilling one of the main GenIV objectives: sustainability.	The activities can shift towards more cross-cutting issues with GenII/III. Transmutation in LWR reactors could be envisaged

No.	Category	Risk denomination	Risk description	Likelihood(*) and Level of criticality of consequences	Possible consequences	Mitigation strategy
7	Modelling	Too high complexity of processes for available modelling tools	The development of physically-based multiscale models is not equally easy in terms of modelling tools and scale bridging for all processes. The models that can be developed might not be sufficient to address the complexity of the processes of interest.	50% Critical in some areas	Lack of suitable models may force to too heavy experimental burden for the full qualification of materials and components.	Hybrid models with a component of empiricism may in some cases prove sufficient.
8	Advanced material solutions	Lack of ODS industrial supplier	The fabrication of ODS fuel cladding tubes is feasible and has been successfully achieved. However, the repeatability and reproducibility of the fabrication procedure at larger scale is still under optimization, in particular the recrystallization behavior of high Cr ODS grades. The production of ODS cladding tubes will remain at lab scale, taking into account that few prototypes will be built in Europe in the next 20-30 years	90% Uncritical	Fuel burn-up will not be increased and the efficiency of the plant will not be optimal, not fulfilling the GenIV requirements.	Alternative materials may be considered: swelling resistant austenitic steels, creep-strength enhanced F/M steels, currently prospective materials, …
9	Advanced material solutions	Qualification as nuclear-grade of innovative materials	Several new alloys or other low TRL material solutions are under explorative R&D to verify if they are as promising as expected, e.g. FeCrAl, AFA, HEA, MAX phases. The long process to become nuclear grade materials involve their qualification in terms of good mechanical properties, oxidation and corrosion resistance, radiation tolerance, compatibility with fuel and neutronic behavior (for fuel cladding) in addition to weldability and formability characteristics. Depending on the material, the possibility that some of the above mentioned characteristics are not fulfilled is high. Also possible that the time for full qualification exceeds the time needs.	50% Critical	No alternative material found to replace those considered in the short term design. Fuel burn-up is not increased, so the efficiency of the plant will remain not optimal. Corrosion mitigation by replacing the existing materials may not be possible. The goals of GenIV systems are not fully achieved.	Design improvements may compensate for the lack of materials.

No.	Category	Risk denomination	Risk description	Likelihood(*) and Level of criticality of consequences	Possible consequences	Mitigation strategy
10	Advanced fuels development	Qualification of advanced fuels	Several new low TRL fuels solutions are under explorative R&D to verify if they are as promising as expected: carbides, nitrides, molten salts Depending on the material, the possibility that some of the necessary characteristics are not fulfilled is high. Also possible that the time for full qualification exceeds the time needs.	50% Severe	No alternative material found to replace those considered in the short term design. Fuel burn-up is not increased, so the efficiency of the plant will remain not optimal. Corrosion mitigation by replacing the existing materials may not be possible. The goals of GenIV systems are not fully achieved.	Design improvements may compensate for the lack of materials.
11	Infrastructures	Availability of nuclear materials exposure and testing infrastructures	Some infrastructures are crucial for the qualification of materials but are either currently not available in Europe (fast spectra), or limited and expensive (MTR, 'hot'cells, loops, …). These infrastructures might end up becoming unsustainable.	50% Can be critical	Insufficient qualification of materials for the design of demonstrators	Set up a scheme of joint planning and management (sharing, exploitation) of nuclear infrastructures in Europe
12	Education and training	Loss of competence	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	50% In some areas critical	Nuclear fission research and competence in Europe may be in danger of decline	Set up a stable joint European scheme of education and training, using materials as a way to attract young people towards nuclear, through collaboration between platforms and coordinated use of funding from projects

No.	Category	Risk denomination	Risk description	Likelihood(*) and Level of criticality of consequences	Possible consequences	Mitigation strategy
13	International cooperation	Lack of exploitation of the potential of cooperation	For international cooperation to be beneficial an effort is needed on the side of international organisations, as well as Euratom, to lay down the proper legal and attractive financial instruments. This may prove to be impossible, either for lack of political willingness and/or financial support	90% Uncritical	Progress worldwide will be hindered (probably with some countries emerging over the others, depending on specific political decisions and resources conceded)	Limited agreements of collaboration, based on opportunities offered by countries or on personal contacts

(*) 10%: Rare (1-20%) - The event may occur only in exceptional circumstances
30%: Unlikely (21-40%) - More likely not to occur under normal conditions
50%: Likely (41-60%) - Given time, likely to occur
70%: Very likely (61-80%) - The event will probably occur in most circumstances
90%: Almost certain (81-99%) - The event is expected to occur in most circumstances

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SUMMARY AND RECOMMEN- DATIONS

12. 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 demonstrators, prototypes and later commercial reactors**, with emphasis on ESNII fast neutron spectrum systems, namely SFR, HLM-cooled systems (ADS and LFR), and GFR. Fast reactors 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, also reducing waste production.

The materials considered here cover as priority the needs of the ESNII demonstrators, but attention is given also to materials solutions that are intended for FOAK and commercial GenIV systems, in which higher energy efficiency and longer burnups than in the demonstrators are targeted, seeking also connections with other GenIV systems, not included in ESNII, i.e. VHTR, SCWR and MSR. The content of this SRA is consistent with other relevant strategic documents and roadmaps compiled by other platforms, 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 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 to define design rules and of the improvement of materials properties; (3) development of innovative material solutions through experimental screening, 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 of interest for other nuclear (current LWR, fusion) and also non-nuclear (e.g. concentrated solar, geothermal, bioenergy, fuel cells and hydrogen, …) energy technologies, with a view to optimising the use of resources, whenever possible, by joining forces with other research communities.

This document also addresses 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, risk assessment.

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 to relevant conditions are the essential ingredient for robust design curves and rules. Plenty of data were produced in the past that are now de facto unusable; this is either because they are covered by confidentiality or because they were not properly archived. Correct data management to guarantee availability for future re-assessment is therefore essential and should be encouraged and fostered. In particular, **financially supported policies to foster data sharing and 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, **pan-European management of infrastructures,** based on **joint programming,** including **trans-national infrastructure renewal planning and a scheme for facility sharing and exploitation**, 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 coordination of activities, sharing of data, and access to infrastructures. Currently, however, **the instruments available in Europe for international cooperation are not sufficiently attractive** to motivate significant cooperation with non-EU researchers. Efforts should be made to improve their attractiveness and ease of access. International **organisations** such as OECD. NEA, IAEA, but also Euratom and JRC for the connection with GIF, have here a crucial role.

R4: The **nuclear materials research community in Europe is currently strongly integrated** and engaged in thriving collaboration, in a bottom-up sense. This is in contrast with the inadequacy of the top-down instruments offered to make this integration efficient and functional. 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 SRA goals should provide the conditions to implement the agreed research agenda and to set up suitable E&T&M schemes that allow 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, a **co-fund instrument, such as a European Joint Programme, seems to be most suitable**.

These recommendations are clearly based on the willingness to pursue a policy of increased integration rather than of isolation at all levels: research organisations, EU Member States, and European Commission. Given 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 the finding of a difficult equilibrium between the need to make the best use possible of the limited resources available, in a framework of nuclear energy where support is politically difficult to obtain, and the legitimate ambition, in a context of healthy competition, to preserve each stakeholders assets.

Joint Programme on Nuclear Materials of the European Energy Research Alliance
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