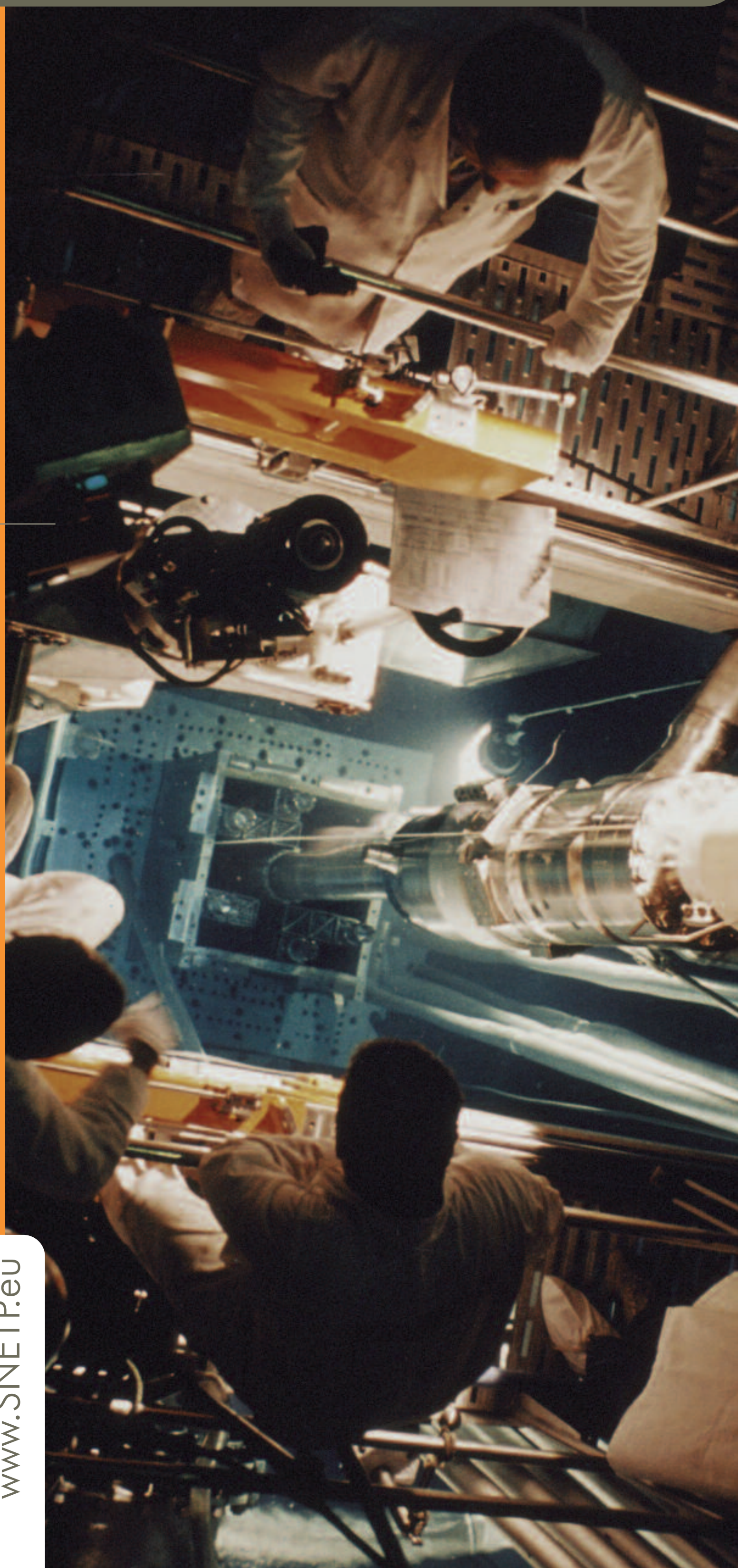
SNETP
Sustainable Nuclear Energy Technology Platform

Strategic Research and Innovation Agenda

February 2013



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SRIA 2013

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The Strategic Research and Innovation Agenda of SNETP is composed of the following documents:

- The Executive Summary Document
- The main Strategic Research and Innovation Agenda Document (this document)

Contact: secretariat@snetp.eu

Foreword

Following its launch in 2007, the first document issued by the Sustainable Nuclear Energy Technology Platform (SNETP) was the Strategic Research Agenda (SRA) now named Strategic Research and Innovation Agenda (SRIA). This document formalised the common vision of more than 70 Member organisations into more accurate research and development orientations and programmes: the SRIA enables the transformation of a shared vision into reality and, thus, the Platform to contribute to European energy policy and to the objectives of the Strategic Energy Technology plan (SET-Plan), in particular with the implementation of the European Sustainable Nuclear Industrial Initiative (ESNII).

Today more than three years after the first edition of the SRIA, SNETP has grown to more than 110 members and has structured its activities: the launching of NUGENIA for the safe and reliable operation of the present reactor fleet (generation II) and the deployment of generation III reactors; implementation of the ESNII initiative which aims to prepare the future deployment of the generation IV nuclear system relying on fast neutron technology with a closed fuel cycle; and the preparation of a future industrial initiative on cogeneration for the supply of heat for industrial purposes.

During this period, the role of nuclear energy remains as strong representing about 30% of the European electricity supply despite its share in the energy mix reducing slightly to 20%. The importance of the role of nuclear energy is acknowledged in the European Commission Communication on the decarbonisation of the economy (Energy roadmap 2050 – COM/2011/885 of 15 December 15, 2011). Electricity is expected to play a greater role in the future energy supply. The scenario studies project a doubling of the electricity share in the energy demand at 40% in 2050, and the same scenarios say that the lowest cost scenarios of decarbonisation are the ones with the highest share of nuclear energy.

But also an important event has occurred during this period with the Fukushima Daïchi accident which has raised concern about the safety of nuclear energy especially in the public mind. It has led to a renewed attention to the safety of nuclear power plants, in particular in respect of extremely severe external hazards. Research and development is an essential tool for better understanding the accident phenomenology and thus for prevention and mitigation of severe accidents. A key responsibility of nuclear operators, and especially in the frame of SNETP, is to take benefit of the lessons learned from this accident. These research and development programmes shall be defined and implemented with the highest priority in the update of the SRIA. In particular, the issue of extremely severe and rare accidents shall be considered in a more global approach to safety in order to better understand the design margins and the behaviour of nuclear reactors under beyond design basis accidents. This will assist the development of more robust measures to prevent and mitigate their possible consequences.

Lastly, this update of the SRIA takes into account the achievements of the R&D programmes described in the 2009 version.

This updated version of the SRIA has been elaborated to respond to this new situation and having in mind the following guidelines:

- **For each part of the SRIA – NUGENIA, ESNII and NC2I – it was crucial to make the role of nuclear safety more explicit as a key driver for the identification of R&D needs.**
- **In line with its expected role in the energy roadmap 2050, nuclear energy is a long term resource for Europe and it will need to enhance its**

sustainability by minimisation of radioactive waste and by optimisation of the use of the available uranium resources while maintaining its competitiveness.

This new edition of the SRIA is compatible and complementary to the SNETP vision report and the original SRA, but more detailed descriptions are provided for the R&D needs to be performed in the short and medium term.

With this updated Strategic Research and Innovation Agenda, SNETP supports the role of nuclear energy in European energy policy for the benefit of its members and European citizens with the highest priority to safety, increased sustainability and consolidated competitiveness for nuclear energy.

Yves Kaluzny

Chairman of the Executive Committee of the Sustainable Nuclear Energy Technology Platform

SNETP

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Executive Summary

www.SNETP.eu

Executive Summary

Introduction

For sustainable prosperity, an affordable and secure energy supply with minimised environmental impact is a primary need, for Europe and beyond. With a growing world population, global energy demand is projected to increase one-third from 2010 to 2035 with an increased share of electricity production. Europe will have to face three energy challenges: security of energy supply, limitation of greenhouse gas emissions and sustained competitiveness of energy-reliant economies.

The European Council committed in March 2007 to very ambitious goals putting Europe at the forefront of the fight against climate change. The "20-20-20" objectives for 2020 are:

- 20% reduction in greenhouse gas emissions compared to 1990
- 20% energy savings
- 20% share of renewable energies in the total energy mix

To achieve both these medium term goals (2020) and a long term vision along this line (2050), the European Commission launched in 2007 the "Strategic Energy Technology (SET) Plan" which identifies a list of competitive low carbon energy technologies to be developed and deployed in Europe. The SET plan identifies nuclear fission as one of the key low carbon energy technologies.

> The primary priority and responsibility in front of European citizens for the nuclear energy sector is, of course, nuclear safety

For the 2020 objectives, the intention is to "maintain the competitiveness in fission technologies together with long term waste management solutions". For the vision of 2050, the SET Plan recommends to act now to "complete the preparation for the demonstration of a new generation of fission reactors with increased sustainability".

The Communication from the European Commission of December 2011, entitled "Energy Roadmap 2050", recognises the importance of nuclear energy's contribution in Europe today. With approximately 122 GWe in operation in Europe, 30% of electricity generation (produced by more than 131 reactors located in 14 countries in the EU-27), nuclear fission represents the largest low-carbon energy source in Europe (2/3 of the decarbonised electricity).

The primary priority and responsibility in front of European citizens for the nuclear energy sector is, of course, nuclear safety. The Fukushima accident increased public concern about nuclear energy and drew renewed attention to the safety of nuclear power plants.

To be sustainable, nuclear energy production must contribute to the well-being of future generations, by reducing the use of natural resources and avoiding detrimental effects on public health and the environment, including the minimisation of ultimate waste.

Security of supply is a key factor in the role of nuclear energy. With the current efficiency of uranium in nuclear power plants and at the projected 2012 rate of consumption, the natural resources may last up to approximately 100 years, depending on the nuclear power growth rate in the next decades. Security of supply will be assured for thousands of years when fast neutron reactors are deployed.

The contribution of nuclear energy to reduce CO_2 emissions could be further increased by using it directly for heat intensive applications. A particular effective approach is using nuclear reactors for cogeneration of electricity and heat.

More than 110 members coming from industry, research & technology organisations, universities, technical safety organisations, service providers, non-governmental organisations and associations have been gathered in The Sustainable Nuclear Energy Technology Platform (SNETP) to define a common vision regarding the role of nuclear energy and R&D needs for the safe, sustainable, and efficient use of nuclear fission technology. SNETP is structured around three main pillars:

- NUGENIA, since its launching in March 2012, is an international association mandated by SNETP. Its main role is to develop R&D supporting safe, reliable, and competitive second (present) and third generation nuclear systems.
- The European Sustainable Nuclear Industrial Initiative (ESNII) was officially launched in November 2010 under the SET Plan. ESNII promotes advanced fast reactors with the objective of resource preservation and minimisation of the burden of radioactive waste.
- The Nuclear Cogeneration Industrial Initiative (NC2I) aims at demonstrating an innovative and competitive energy solution for the low-carbon cogeneration of process heat and electricity based on nuclear energy.

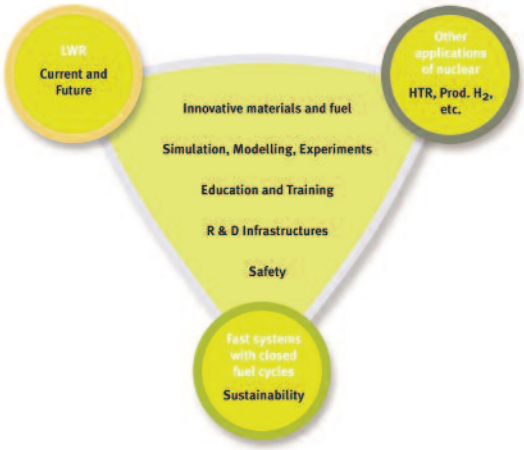

Figure 1: The SNETP three pillars

Safety vision

The safety of nuclear installations results from a permanent process of improvement to both reduce the probability of accidents and mitigate their consequences and requires dedicated research and development. In the past, the bulk of safety research has been mostly carried out within national programmes supported either by public financing schemes or by the operators. More recently, the R&D effort is however being increasingly shared internationally, in particular through the EURATOM Framework Programmes and the OECD/NEA programmes. Thanks to the work performed within SNETP, the present release of the SNETP Strategic Research and Innovation Agenda is able to present a more comprehensive list of the issues that Platform members acknowledge ranking at highest priority. Most of these topics are also suitable for harmonisation at European level.

The Fukushima accident has increased society's concern for the safety of nuclear power plants. Although the detailed analysis of the accident will take many more years, it will result in an increased emphasis on particular R&D. In the framework of this enhanced global effort, the SNETP Governing Board empowered a Task Group to investigate how the lessons learned from the Fukushima accident could impact safety related R&D orientations and priorities. According to the conclusions of the Task Group, no really new phenomena were revealed by the Fukushima accident, but the Task Group identified 13 main research subjects to be addressed with the appropriate priority, in the areas of plant design and identification of external hazards, analysis and management of severe accidents (in particular new systems for mitigation of their consequences), emergency management and radiological impact.

In parallel, most of the countries operating nuclear reactors have launched systematic reassessments of the safety margins of their nuclear fleet under exposure to severe natural hazards. The European Council of 24-25 March 2011 requested that a comprehensive safety assessment be performed on all EU nuclear plants, in the light of the preliminary lessons learned. The request of the Council comprised “stress tests” performed at national level, complemented by a European peer review. Although the stress tests were based on information available at the time and were not primarily intended to specify areas for future research, they also indicated the need for future studies and developments.

Looking ahead, and considering in particular the outcomes from the Fukushima accident, the regulators - according to the WENRA objectives - will require enhanced safety for reactors already in operation as well as so for those either in construction or to be built. To address these new and more demanding requests, pre-normative research is to be promoted within SNETP to achieve the inherent safety objectives. More generally, research should contribute to allow "best estimate" evaluation of the reactor systems' behaviour, up to the cliff edge threshold resulting in their complete degradation.

More inputs from the analysis of the Fukushima event will be generated in the coming years both from the analysis of the situation on the site and from the application of simulation codes to the accident. It can be, however, already confirmed that the main challenges identified from the lessons learned from the accident are the following:

- To better characterise any natural events, like earthquakes, floods, etc., including methodologies for dealing with rare events.
- To extend even more in-depth the safety response to any type of initiating event, especially severe natural hazards and any combination of them. It shall be done for current reactors, Generation III reactors and future reactors.
- To include more systematically at the design stage the beyond design basis accidents to assure the robustness of the defence in-depth and to avoid cliff edge effects. The approach shall include situations where all units on the same site are affected by a beyond design event.
- To develop wider and more robust lines of defence with respect to design basis aggressions and beyond design basis events, by defining additional measures to consider in the design and new or improved systems for mitigation of consequences.

It is therefore expected that safety analysis of the nuclear installations will include in the future, additional advanced elements, like:

- Evaluation of the best estimate behaviour of the nuclear installation systems for beyond design basis accidents to assess possible challenges to the fulfilment of safety functions.
- Evaluation of the ultimate capacity of the systems with respect to the load applied and precise identification of the margins and provisions that prevent non-linear or catastrophic damage.

This will require an extension of the capability of physical modelling and computer tools in different areas and in particular in the area of severe accident and containment system simulation, enhanced in order to derive the radioactive source term due to the accident. Experimental and theoretical research efforts will be necessary to support the possible evolution of the safety regulations and practices.

Specific emphasis will have to be put also on emergency management, which has been challenged during the Fukushima accident due to:

- the concomitance of many events, the severe environmental conditions and the mutual interaction between the affected units on site
- the complexity and the difficulty of the decision making process which diminished the effectiveness and promptness of actions and which generated both confusion and delays
- the practical impossibility of recovering a workable and stable electrical supply source for several days

The improvement of emergency preparedness and response shall include the consideration of several items:

- the availability of more sophisticated tools to provide to the operators with more reliable and quick indications/measurements of reactor status, to help in the implementation of an appropriate recovery strategy
- the availability of redundant intervention means in the vicinity of the site
- the availability of better and faster environmental monitoring systems, better models for contamination predictions, health effects of low doses, and effect of contamination on the environment – in particular the marine environment
- a broader consideration of organisation and social issues and cultural aspects
- better international cooperation/expertise which could provide help with plant status diagnostics, with forecasting accident evolution and on mitigation strategies

Sustainability of the Nuclear Fuel Cycle

A large number of studies have been carried out worldwide, and particularly in Europe but also within the framework of the Generation IV roadmap, to analyse the meaning of "sustainability" when it is applied to the NFC. From that work, there is clear consensus today that a sustainable NFC is mainly linked to the durability of the solutions addressing the three following issues:

- optimum use of natural resources
- nuclear waste minimisation
- minimum impact on environment

These objectives must be pursued while maintaining or increasing at the same time the safety, the economic competitiveness and the protection against diversion or undeclared nuclear material production and misuse of technology.

A higher burn-up indirectly leads to a net reduction of natural uranium consumption

The change to enhanced sustainability is a progressive process that has already started. As a matter of fact, some of the technologies for plutonium-recycling fuels are commercially available and industrially operational, in particular in some EU countries, e.g. France, UK.

One of the most efficient routes to reducing natural uranium consumption is to increase the conversion ratio (ratio between the total amount of artificial fissile material created inside the reactor core and the total amount of fissile isotopes "consumed") of present and future reactors and to recycle fissile material.

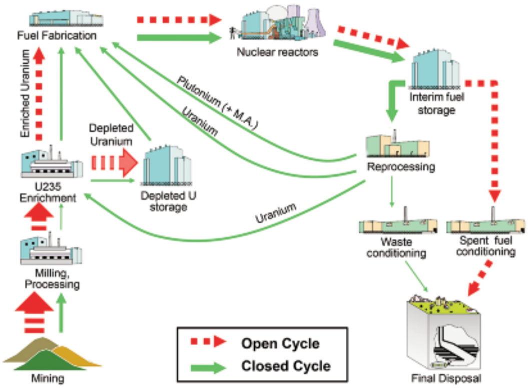

Figure 2: The closed nuclear fuel cycle illustrating the recycling and optimisation of energy resources and the minimisation of nuclear waste (Source: CIEMAT)

First improvements in the use of natural resources could be made through advanced LWR systems with higher conversion ratio and improved fuel design, and the associated back end of the fuel cycle. Attractive opportunities have already been identified for Advanced Generation III reactors, such as new core designs and loading strategies, fuels accommodating very high burn-ups for LWR (>70 GWd/tHM) and improved capacity for plutonium and reprocessed uranium recycling, including 100 % MOX cores, plutonium multi-recycling for LWR and 100% plutonium cores for HTR. The corresponding R&D challenges have been taken into account in the NUGENIA chapter.

However, fast nuclear reactors can be designed to reach conversion ratios equal or even greater than one, in such a way that no more natural fissile isotope is needed to sustain nuclear energy since the reactors generate more fissile isotopes than they consume to produce energy. These reactors, called "breeders" need to be fed at equilibrium with fertile isotopes (^{238}U or ^{232}Th) which are available in plentiful amounts, both in nature and as leftovers from the present enrichment of the nuclear fuel in ^{235}U.

High level waste (HLW), which contains highly radioactive isotopes, significant quantities of long lived radio-nuclides and is strongly heat emitting, is mainly generated by the operation of nuclear reactors. HLW can be the spent fuel or waste from its reprocessing or from other steps of the NFC. The present solution for HLW is to condition it inside isolating and protecting packages that are then disposed of in a Deep Underground Geological Repository (DGR). A number of technological and geological barriers are set up in this way to avoid any hazard to the population or the biosphere. This solution has been scientifically proven to be reliable and safe, and most of its technologies are ready for deployment. The first implementations in the EU are expected in Finland, Sweden and France within the next 7-20 years.

To optimise HLW management, research should focus on minimising several parameters of the HLW: the mass and volume of conditioned NW to be disposed of, the long term radiotoxic inventory, the effective "lifetime" of conditioned NW; the heat generation of conditioned NW as function of time and the long term radiological impact.

These objectives can be achieved conceptually in two generic types of scenario:

- A fleet of fast neutron critical reactors that simultaneously produce electricity and transmute all the actinides. The only input into the system (reactors and fuel cycle facilities) is natural or depleted uranium and the output is electricity and residual Intermediate Level Waste (ILW) plus HLW, including the fission fragments, activation products and actinide reprocessing losses. In this option, the minor actinides (MA) could be homogeneously diluted within the whole fuel or separated in the form of dedicated targets.

However the core design of these reactors has to be optimised from the point of view of neutron economy and safety performance, and the feasibility of the associated fuel cycles should also be addressed.

- A "double strata" reactor fleet. The first stratum is a set of critical reactors dedicated to electricity production using "clean fuel" containing only U and Pu. The reactors in this stratum can be either present or future thermal reactors or fast reactors, or an appropriate combination of both generations. The second stratum is devoted to transuranic elements (TRU) or MA transmutation and is based on special fast reactors or subcritical fast systems, Accelerator Driven Systems (ADS), loaded with homogeneous fuels with high MA content.

The evaluation of this type of scenario indicates that while maintaining the safety of operation, they should ultimately be able to significantly reduce the long term uranium consumption, making the present reserves last for several thousand years. At the same time, the HLW long term radiotoxic inventory could be reduced by more than a factor of 100 and its heat load by more than a factor of 10, at medium and long term. According to these studies the last figure will allow the DGR capacity to be increased by factors from 3 to more than 10 (in hard rock, clay and tuff geological formations).

The deployment of these advanced fuel cycles involves large technological challenges:

- new fuels (targets) and fuel assembly designs bearing significant amounts of MA, and their fabrication technology
- the technologies of FNR and ADS, including new materials, thermal-hydraulics, simulation tools, nuclear data and, in the case of ADS, the coupling of an accelerator with a subcritical core
- new recycling technologies based on advanced aqueous and pyro-metallurgic reprocessing, adapted to highly active and hot fuels containing large amounts of Pu and MA, and minimising the production of secondary wastes

Additional fuel cycle scenarios studies are required to complete the evaluation on the feasibility of sustainable solutions for the transition period from the present nuclear fleet until the deployment of fast nuclear systems, taking into account present perspectives for deployments of advanced thermal reactors and future FNRs. Similarly, the evaluation of the impact of these technologies in the DGR designs, taking into account updated nuclear policies of EU Member States, technology deployment and different options for the fast systems deployments, needs still to be completed.

Finally, it should be mentioned that, although currently there are no short or medium term industrial prospects in Europe for the deployment of the thorium cycle, thorium could become an attractive option for the long term due to its large European resource base and potential role in the nuclear waste minimisation. An interesting strategy for the long term could be the combination of Molten Salt Reactors (MSR) technologies. Both thermal and fast neutrons MSR with the thorium fuel cycle could become important long term research topics.

NUGENIA - nuclear fission technologies for Generation II and III nuclear plants

NUGENIA (NUclear GENeration II & III Association), established on 14 November 2011, developed the roadmap which forms the basis for the Generation II & III part of the SNETP SRIA. The main mission of NUGENIA, which received a mandate from SNETP, is to be the integrated framework between industry, research and safety organisations for safe, reliable and competitive Generation II & III nuclear fission.

Figure 3: Golfech NPP over the Garonne river (Source: EDF)

The R&D under NUGENIA is organised in six technical areas:

1. Plant safety and risk assessment
2. Severe accidents
3. Improved reactor operation
4. Integrity assessment of systems, structures and components
5. Fuel, waste management and dismantling
6. Innovative LWR Generation III design

plus two cross-cutting areas:

- Harmonisation
- In-service inspection and inspection qualification

Each area coordinates its detailed roadmap while ensuring proper transverse homogeneity. Periodic updates of the roadmap (typically every 3-4 years) will allow adaption to evolving contexts. The roadmap presented in this SRIA is based on extracts from the detailed NUGENIA roadmap.

NUGENIA Technical Area 1 (TA1), Safety and risk of NPPs, is devoted to improving understanding and numerical representation for the relevant phenomena involved in incidents and accidents at NPPs, in order to increase the realism of plant behaviour assessment and to enhance the accuracy of safety margin assessment. The main challenges identified within this TA1 are:

NUGENIA Technical Area 1 (TA1) deals with Safety and risk of current Nuclear Power Plants

- challenges in the field of Probabilistic Safety Assessment, including quantitative aspects, methodologies to assess shut-down state, and external events, assessing of risk related to spent fuel pool, and best practice for probabilistic safety assessment (PSA) application.
- deterministic assessment of plant transients: improving models for plant transients including thermal hydraulics, design and evaluation of passive safety systems, coupled multi-physics codes, containment behaviour, fluid structure interactions.
- impact of external loads (including electrical disturbances) and other hazards on the safety functions
- advanced safety assessment methodologies: safety margins and best estimate methods, integrating the deterministic and probabilistic safety assessments,
- design of new reactor safety systems.

NUGENIA Technical Area 2 (TA2) is devoted to severe accidents. Despite the highly efficient accident prevention measures adopted for the current Generation II and the still more demanding ones for the Generation III plants, some accident scenarios may, with a low probability, result in a severe accident (SA), as recently enphasised with the Fukushima Daiichi accident in Japan. This SA can eventually result in core melting, plant damage and dispersal of radioactive materials outside the plant containment, thus threatening public health and the environment.

NUGENIA Technical Area 2 (TA2) is devoted to severe accidents

This risk can be substantially decreased when state-of-the-art devices currently available for prevention and mitigation of severe accidents are installed. Lessons from the Fukushima accident and consequences related to accident management provisions from the recently completed ENSREG stress tests and other national activities will lead to further enhancement of the safety of NPPs.

Within NUGENIA TA2, general objectives are defined and followed to further reinforce NPP safety provisions through better understanding of some predominant phenomena, improving Severe Accident Management Guidelines (SAMGs) and designing new prevention devices or systems for mitigation of SA consequences.

The highest priority safety challenges are described in the following sub-areas:

- in-vessel corium/debris coolability
- ex-vessel corium interactions and coolability
- containment behaviour including hydrogen explosion risk
- source term
- impact of severe accidents on the environment
- severe accident scenarios
- emergency preparedness and response

NUGENIA Technical Area 3 (TA3) is devoted to improved reactor operation. Safe and efficient operation of the plants is the result of a blend of human, organisational and technological aspects. The R&D topics developed to improve reactor operation include the following issues:

- human and organisational factors
- integration of digital technologies
- core management
- water chemistry and LLW management
- radiation protection

After the accident at Fukushima, studies on Human and Organisational Factors (HOF) will be mainly addressing human and organisational performance in emergency conditions, to support preparedness for an emergency.

NUGENIA Technical Area 3 (TA3) is devoted to improved reactor operation

Digital technologies are nowadays deployed in all modern power generation plants and also in large industrial plants. In the nuclear power sector, however, the regulatory conditions and financial risk are favouring extending the use of analogue systems, even beyond their initially expected service lifetime, and delaying the deployment of new technologies. Implementation of digital technologies is becoming a more and more urgent issue for the life extension of the Generation II reactors, as well as for the deployment of the Generation III, offering a unique opportunity for improving operational performance while respecting safety margins.

Other operational targets supported by research in TA3 will be optimisation of core loading strategies, water chemistry management and reduction of radiological dose for workers.

NUGENIA Technical Area 4 (TA4) addresses Systems, Structures and Components (SSCs). Structural assessments of SSCs are an important part of NPP management programmes (e.g. ageing management, maintenance and design changes) to improve safety and availability. These assessments are required, for instance, to enable and improve the effectiveness of periodic safety reviews. Aspects that need to be considered include definition of integrity assessment over the whole life cycle, the various degradation mechanisms, ageing issues, safety margins and harmonisation. While the assessment principles used in that context are generally comparable in Europe, the actual methodologies and codes are still different in the various European countries.

NUGENIA Technical Area 4 (TA4) addresses Systems, Structures and Components (SSCs)

The objective of NUGENIA TA4 is to improve understanding, and to develop methods and tools in order to increase the safety and availability of systems, structures and components needed for reliable and safe management of nuclear power plant lifetime. Research challenges requiring R&D programmes have been identified as follows:

- integrity Assessment, both for metallic components and concrete
- quantification of the loadings, also for metallic components and concrete
- materials Performance and Ageing, covering material properties, metallic components, polymer materials, ageing and degradation mechanisms, metallic component issues, issues for concrete structures, polymer materials and modelling of ageing
- ageing Monitoring, Prevention and Mitigation, including topics on ageing monitoring of metallic components, R&D topics on concrete material, polymers and electrical equipment, and prevention and mitigation of ageing for metallic components and concrete
- functionality, including R&D topics on equipment reliability, industrial obsolescence, maintenance and qualification

NUGENIA Technical Area 5 (TA5), Fuel Development, Waste and Spent Fuel Management and Decommissioning, covers development of nuclear fuel for existing, advanced and innovative core designs, aspects of fuel use in reactors (nuclear fuel behaviour mechanisms) and the fuel management steps – manipulation, transport and interim wet and dry storage. It also includes factors relating to the generation and management of radioactive waste, and the dismantling and decommissioning of nuclear power plants.

NUGENIA Technical Area 5 (TA5), Fuel Development, Waste and Spent Fuel Management and Decommissioning

Nuclear fuel production and use has reached a relatively mature state; nevertheless there is motivation to improve LWR oxide-based fuel types; to increase burn-ups; to increase fuel reliability safety margins; to reduce reactor operating costs (including fuel costs); to reduce the amount and/or radio-toxicity of spent fuel; to recycle existing waste (uranium, plutonium and minor actinides from prior reprocessing operations); to improve proliferation resistance.

In order to perform this research it is necessary to maintain key experimental infrastructures, such as flexible high flux irradiation facilities, hot cells and PIE laboratories.

NUGENIA Technical Area 6 (TA6) is devoted to innovative LWR design and technology

Decommissioning R&D will focus on waste minimisation strategies and on the development of efficient dismantling technologies for structures and components, including remote dismantling techniques.

NUGENIA Technical Area 6 (TA6), Innovative LWR design. In advance of industrial deployment of the fourth generation of nuclear reactors (ESNII), and considering the ageing of the current European nuclear power plant fleet, there will be an opportunity for preparing the next Light Water Reactor generation for electricity production throughout the 21st century.

R&D for the design and construction of reactor components will be a cross-cutting aspect that will apply to all Light Water Reactors, existing and new designs, in order to improve: safety & commissioning, operability, sustainability, economics and public acceptance especially following the Fukushima accident.

Knowing that new technology deployment at the industrial scale could be a long process, the following time lines will be considered:

- evolutionary technology for mid-term application
- breakthrough technology for the longer term
- advanced LWR designs such as with higher conversion ratio or small modular reactors, expected to be ready for commercial operation by 15 to 20 years

The R&D work proposed to support existing and new light water reactor concepts, will be focused on achieving long term operation by design; safety by design; innovative components for reduced maintenance; and enhanced economics.

Harmonisation is aimed at reducing any substantial difference within a group of countries in design and fabrication of systems and components, in nuclear safety level and requirements, as well as in safety assessment processes and practices. It involves the search for a long-term convergence towards the agreed WENRA objectives.

Accordingly the objectives of the *NUGENIA harmonisation Technical Area* are setting up the R&D basis for an effective standardisation of reactor component assessments and improving the safety level of the nuclear installation through shared design approaches and licensing processes.

There are three main fields of endeavour:

- pre-normative research (PNR) for new design and operating conditions, but also for definition of limits, criteria and establishment of practices
- establishment of shared codes and standards
- strategy providing smooth and efficient methods to progressively enlarge consensus among stakeholders

For the development of these fast reactors within ESNII, it is of paramount importance to excel in safety, reliability, radiological protection and security

Challenges for inspection and qualification will be the qualification of procedures based on new non-destructive testing technologies, the optimisation of in service inspection frequency, to be based on risk reduction quantification, and the definition of methodologies for pre-service inspection for new builds.

European Sustainable Nuclear Industrial Initiative ESNII

One of the major concerns of society with regard to the implementation of nuclear energy is the high-level nuclear waste. Fast spectrum reactors with closed fuel cycles will allow a significant reduction in high-level nuclear waste radiotoxicity and volume. Fast reactors will also allow an increase in natural resource (uranium) utilisation by a factor of around 50. In this way, it is clear that the use of fast reactors with a closed fuel cycle approach will allow more sustainable implementation of nuclear energy.

For the development of these fast reactors within ESNII, it is of paramount importance to excel in safety, reliability, radiological protection and security.

The main objective of ESNII is to maintain European leadership in fast spectrum reactor technologies that will excel in safety and will be able to achieve a more sustainable development of nuclear energy. With respect to the 2010 evaluation of technologies, sodium is still considered to be the reference technology since it has more substantial technological and reactor operations feed-back. The Lead(-bismuth) Fast Reactor technology has significantly extended its technological base and can be considered as the shorter-term alternative technology, whereas the Gas Fast Reactor technology has to be considered as a longer-term alternative option. **The main goal of ESNII is to design, license, construct, commission and put into operation before 2025 the Sodium Fast Reactor Prototype reactor called ASTRID and the flexible fast spectrum irradiation facility MYRRHA.**

ASTRID will allow Europe to demonstrate its capability to master the mature sodium technology with improved safety characteristics responding to society's concern of having the highest possible level of safety. Therefore, the design of ASTRID focuses on meeting the challenges in terms of industrial performance and availability, improved waste management and resource utilisation and a safety level compatible with WENRA objectives for new nuclear build, whilst at the same time targeting to achieve of the Generation IV goals. An associated R&D programme will continue to accompany and support the development of ASTRID to increase the lines of defence and robustness of this technology, and allow the goals of the 4th generation to be reached, not only on safety and proliferation resistance, but also on economy and sustainability.

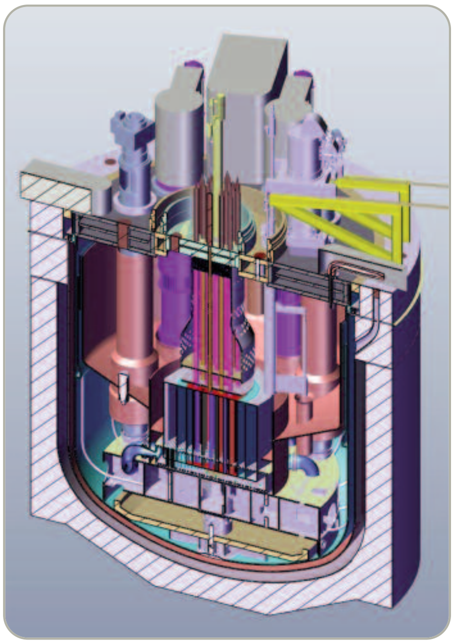

Figure 4: ASTRID design primary system (Source: CEA)

With MYRRHA, Europe will again operate a flexible fast spectrum irradiation facility in support of the technology development (in particular for material, components and fuel irradiation tests) of the three fast reactor systems (SFR, LFR and GFR). Also, MYRRHA will offer a wide range of interesting irradiation conditions for fusion material research. Since MYRRHA will be conceived as an Accelerator Driven System, it will be able to demonstrate the ADS technology, thereby allowing the technical feasibility of one of the key components in the double strata strategy for high-level waste transmutation to be evaluated. Due to the fact that MYRRHA will be based on heavy liquid metal technology (namely lead-bismuth eutectic), it can serve the role of Lead Fast Reactor European Technology Pilot Plant (ETPP) as identified in the LFR roadmap. An associated R&D programme will accompany and support the development of MYRRHA.

For the financing of the total investment cost of these facilities, it will be of paramount importance to establish the appropriate consortium structure and legal basis, allowing candidate consortium members to identify the added value of the facility for their own interest.

In parallel to the realisation of ASTRID and MYRRHA, activities around the Lead Fast Reactor technology and the Gas Fast Reactor technology should be continued, taking into account their specific needs.

For the development of the Lead-cooled Fast Reactor, maximum synergy of activities will be sought with the MYRRHA development to optimise resources and planning. For the LFR demonstrator ALFRED, the main focus should be on design activities typical for a critical power reactor connected to the grid, as well as on R&D activities on the lead coolant, addressing the specific characteristics that differ from lead bismuth. Design activities and support R&D shall be performed in the next years to the maximum extent compatible with available resources and taking full advantage of feedbacks, where applicable, from the ongoing design of MYRRHA and related R&D programmes. These activities will allow the LFR consortium to reach the level of maturity needed to start the licensing phase and then the construction of ALFRED, provided that adequate financial resources are made available.

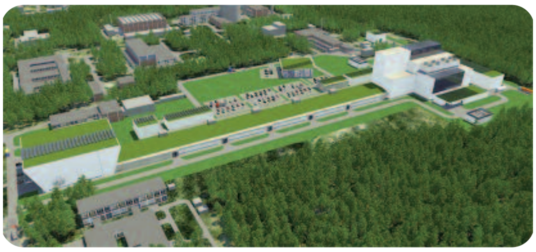

Figure 5: MYRRHA Lay-out picture (Source: SCK-CEN)

In addition to the closure of the nuclear fuel cycle in a sustainable manner, the Gas Fast Reactor has the potential to deliver high temperature heat at ~800 °C for process heat, production of hydrogen, synthetic fuels, etc.. The Helium cooled Fast Reactor is an innovative nuclear system having attractive features: helium is transparent to neutrons and is chemically inert. Its viability is however essentially based on two main challenges. First, the development and qualification of an innovative fuel type that can withstand the irradiation, temperature and pressure conditions put forward for the GFR concept. Secondly, a high intrinsic safety level will need to be demonstrated for this GFR concept. This will imply dedicated design activities followed probably by out-of-pile demonstration experiments. These high priority R&D activities should be embedded into an overall R&D roadmap in support of the development of the Gas Fast Reactor concept. For the development, guidance and implementation of this R&D effort, a GFR centre of excellence will be created. This centre could develop the technical capability to launch the ALLEGRO gas cooled demonstrator.

Based on the ADRIANA project, a number of supporting facilities for the different systems and technologies have been identified. The realisation and operation of these supporting facilities, in particular a fast reactor MOX production line, will be of primary importance to reach the aforementioned objectives.

Raising the financial resources to carry out the ESNII projects and to build the different facilities will be a key factor of success. In this respect, international collaboration through GIF and bilateral or multi-lateral frameworks will be sought to optimise resources. In the next years, project financing capabilities may modify the ESNII part of this Strategic Research & Innovation Agenda.

Nuclear Cogeneration

Nuclear cogeneration relates to the co-production of heat and electricity using a nuclear reactor. Fossil fuels are today by far the main source of heat for European industry, transport and households. The production of heat with nuclear technology is a major innovation that can open a new and significant market potential for nuclear systems, whilst providing a notable contribution to European energy policy in terms of curbing CO_2 emissions and increasing security of energy supply.

Additionally, short-term opportunities for households such as district heating, and desalination to solve fresh water shortages, and long-term opportunities for reducing fossil fuel usage in transport, by generating synthetic fuels via nuclear powered hydrogen production, add a significant market and carbon emission reduction potential.

Nuclear cogeneration is already a reality. In Europe, more than 1000 GWh of low-temperature heat was produced in 2006 in Bulgaria, Czech Republic, Hungary, Romania, Slovakia and Switzerland. Water reactors have extensive operational experience, including in low-temperature cogeneration. Low temperature cogeneration from a fast neutron reactor was proven for desalination in the case of one Kazakh plant (BN-350). Significant development is however needed before nuclear cogeneration can be considered for medium level temperature applications.

High temperature reactors (HTR) on the other hand provide significant perspectives for medium and high temperature cogeneration applications. The HTR features high efficiency, due to elevated primary coolant temperatures, and a very high level of inherent safety. The HTR technology builds on the developments in Germany in the 1980s, as well as in previous research in UK and USA, re-established and revived in several national and European framework programme projects from the year 2000 onwards. The coupling of HTR with end-users for high temperature cogeneration has still to be developed.

The HTR design allows high flexibility in terms of power rating and temperature

Two HTR design concepts use the same high safety standard TRISO fuel particles either embedded in graphite spheres for the pebble bed core or in compacts inserted into graphite blocks in the block-type core. The present major realisations of the design are the test reactors HTR10 (pebble bed) in China and the HTTR (block-type) in Japan. The HTR design allows high flexibility in terms of power rating and temperature. In addition, its inherent safety characteristics, including limitation of fuel temperature in case of accidents and the threefold containment of radioactivity in the TRISO fuel particles, complemented by the inert helium coolant, underscore the high safety credentials of HTR.

For HTR-development, the design options can be classified into:

1. **Short term**: indirect cycle, steam production 550-600°C (current coal fired power plant conditions as reference), power/heat split depending on demand
2. **Mid term**: follow materials development towards higher temperature applications in fossil-fired plants, possibly switch to a heat carrier other than steam
3. **Long term**: 950°C or beyond (primary side) requires change and thus development of structural material for applications such as thermo-chemical H2 production and other high temperature processes

The R&D efforts for nuclear cogeneration implementation can be subdivided into three parts: generic R&D, R&D towards demonstration of nuclear cogeneration using high temperature reactor technology and R&D to broaden the potential of HTR technology.

The following nuclear cogeneration R&D subjects are generic, as they are relevant for all nuclear systems operating in cogeneration mode, but their importance and relevance depend strongly on the nuclear system envisaged, and the process it is connected to:

- tritium transport reduction to secondary and tertiary systems
- impact of process transients on cogeneration supply unit, and vice versa
- coupling technology including energy buffering
- adaptation of existing LWR and future SMR to meet strongly growing demand for district cooling and seawater desalination in arid countries

R&D towards demonstration of nuclear cogeneration using high temperature reactors mainly concerns safety demonstration, licensing support and technology innovation to maintain and strengthen the HTR knowledge base and support demonstration, possibly through international collaboration.

Beyond demonstration, the potential of nuclear heat sources can be further broadened by appropriate R&D. Of particular relevance is the research in the following areas:

- for HTR: increased primary coolant temperature (950°C) for enhanced efficiency and broader application perspectives
- alternative fuel cycles, including thorium, to conserve fuel resources, minimise waste and optimise cycle length

Major investments are needed in modern experimental infrastructure and facilities to enable the above R&D to be performed adequately. The following facilities are essential:

- new irradiation facilities for the investigation, characterisation, development and, ultimately, validation and qualification of HTR fuels and materials. Accident testing requires development of novel and specific irradiation test facilities and equipment. Additionally an in-pile helium loop is needed to assess material behaviour under representative primary coolant flow and irradiation conditions

- out-of-pile testing facilities such as accident test helium loops
- modern hot cells with heating tests and state-of-the-art PIE possibilities to enable the generation of the appropriate data for code development and validation, and to increase the fundamental understanding of material and fuel behaviour
- fuel manufacturing laboratory, also able to handle transuranic elements

These facilities serve two purposes: on the one hand they are essential for appropriate R&D to be performed, on the other, they form the basis for the design, licensing and operation of a demonstration plant.

Cross-cutting R&D topics

In the present SRIA the R&D topics have been organised according to the related reactor technologies, however some topics have intrinsic cross-cutting nature. This is the case for fuel cycle technologies bridging different reactor generations, in particular the fuel reprocessing. Education & Training and Knowledge management are also topics affecting to all nuclear technologies. Knowledge management is essential as it allows storing and disseminating the results of research.

Many opportunities to improve the optimal utilisation of natural resources and nuclear waste minimisation are open by the reprocessing of the nuclear fuel after its use in nuclear reactors. The fuel reprocessing allows separating materials that can be reused in thermal or fast nuclear reactors, either to produce additional energy or to minimise the final waste to be sent to the geological repository. Indeed, the reprocessing of used nuclear fuel is a critical component of all the strategies for long-term sustainability of nuclear energy.

Reprocessing of the fuel used in the present LWRs is a common industrial practice in France, and similar technologies also available in the UK. The plutonium and uranium recovered are partially recycled in the same LWRs in the form of MOX and the rest is saved for use in future FNR.

The challenges for the R&D in fuel reprocessing, include the industrialisation of laboratory technologies for separation of minor actinides from the high level wastes of the reprocessing of the fuel used in the present reactors; the development of reprocessing of advanced fuels foreseen for future reactors (FNR, ADS, advanced thermal reactors and HTR); technologies able to perform joint extraction of several actinides; and the minimisation of secondary wastes in all these strategies. These developments should be performed coherently with the technologies for advanced fuel fabrication and characterisation.

With regards to E&T the key challenge is still to raise the attractiveness for qualified young people of studies and professions related to nuclear technologies

In the short term, the required R&D for nuclear waste reprocessing can be performed in several existing basic science and validation facilities, but in the medium term demonstration plants for the reactors, fuel fabrication and advanced reprocessing technologies will be needed, both at national and European levels. At long term the R&D should focus on the industrial implementation of partitioning and transmutation.

With regards to E&T the key challenge is still to raise the attractiveness for qualified young people of studies and professions related to nuclear technologies. This challenge has been enhanced by the impact of the Fukushima accident. Systematic approaches are under preparation to develop solutions tailored to meet the challenges that nuclear E&T is facing in the near future.

Of particular relevance is the relationship between nuclear education and training and nuclear research: First, the quality of nuclear research directly depends on the interest and engagement of highly qualified scientists and engineers in those activities. Second, research plays a crucial role for the qualification of young scientists and engineers by providing know-why and other important competences required to solve relevant technological and safety issues and to ensure the capability for leadership in organisations involved with nuclear energy issues.

SNETP
Sustainable Nuclear Energy Technology Platform

Strategic Research and Innovation Agenda

February 2013

SNETP
SRIA 2013

www.SNETP.eu

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Message from the SNETP Chairman

Launched in September 2007, the Sustainable Nuclear Energy Technology Platform (SNETP) gathers more than 110 members from at least 22 European countries. The SNETP issued the first Strategic Research Agenda in 2009 defining three pillars of the platform. These pillars evolved into functional working bodies with a defined European outlook and identified representatives.

The first pillar integrates and develops further R&D capabilities to maintain the safety and competitiveness of existing technologies. SNETP has mandated the international association NUGENIA established in March 2012 to take over the work originated in the SNETP WG Gen II/III, NULIFE, and SARNET.

In the second pillar are gathered actors with the aim of developing a new generation of more sustainable reactor technologies. ESNII was launched as an industrial initiative recognised by the SET-Plan in September 2010.

Since 2011 the third pillar, focused on developing new industrial applications of nuclear power, is represented by NC2I Task Force.

The period 2007-2013 has witnessed the consolidation of SNETP within the European framework, as well as the defining of priorities and deployment strategies. The publication of an updated Strategic Research and Innovation Agenda (SRIA) allows the platform to update after 4 years since the first issue the priorities and strategic approaches. The content of the second edition of the SRIA has been open to broad consultation and represents the viewpoint of various stakeholders representing utilities, vendors, technology providers, research organisations, technical safety organisations, universities, consultancy companies and non-governmental organisations.

I would like to express my gratitude to all of them.

F. Pazdera
Chairman of the Sustainable Nuclear Energy Technology Platform

1. Introduction

For sustainable prosperity, an affordable and secure energy supply with minimised environmental impact is a primary need, for Europe and the rest of the world. With a growing world population, global energy demand will increase one-third from 2010 to 2035[1].

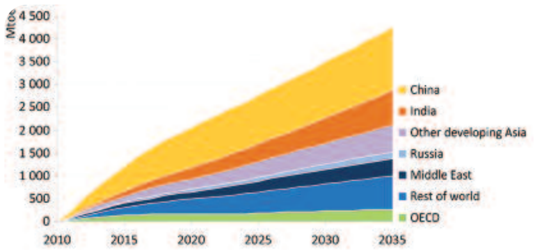

Figure 6: Increase of Energy Consumption Worldwide 2010-2035 (Source: IAEA World Energy Outlook 2011)

Currently the major share of energy needs is covered by fossil fuel resources; consequently increased geopolitical tensions and energy price volatility have a negative impact on the economy. Energy security has become a global concern. The target to decarbonise Europe's economy by 80-95% by 2050 implies a major reduction in greenhouse gas emissions. Moreover, an affordable, secure and sustainable energy supply is necessary to preserve prosperity in Europe. The International Energy Agency (IEA) has highlighted the critical role of governments and underlined the need for urgent action.

The Sustainable Nuclear Energy Technology Platform is the European Technology Platform gathering stakeholders involved in the research and innovation and in the demonstration and deployment of nuclear fission reactors and fuel cycle facilities, and the associated education and training. More than 110 members from industry, research & technology organisations, universities, technical safety organisations, service providers, non-governmental organisations and associations share a common vision on the role of nuclear energy and on the need for a safe, sustainable, and efficient use of nuclear fission technology.

SNETP aims at promoting the research, development and demonstration of European nuclear fission technologies.

This second edition of the Strategic Research Agenda, issued four years after the first one, is now called Strategic Research and Innovation Agenda (SRIA), to underline the importance of Innovation in the framework of the nuclear fission technology road-map. It takes into consideration changes occurring in recent years, both due to the economic crisis, and due to the Fukushima accident and its implications at political level. It also takes into account that development of nuclear technology should be done in a progressive way, to warranty that safety is respected and the very large investments involved are protected, along the whole process from conception to demonstration.

SNETP aims at promoting the research, development and demonstration of European nuclear fission technologies

This issue includes an assessment of the achievements and progress since the previous release, an evaluation of the lessons learnt from the Fukushima accident (including conclusions from the SNETP Fukushima Task Force) and an update of the present SNETP vision of nuclear research, making more explicit, both, the role of safety on all points of the nuclear R&D programmes, and the principle of enhanced sustainability by the minimisation of waste and optimisation of the use of available resources, while maintaining competitiveness.

Over 100 scientists, researchers and engineers have contributed to the update of this document. NUGENIA and the rest of the Working Groups in SNETP have collaborated in the creation of the Strategic Research and Innovation Agenda.

1 - IAEA World Energy Outlook 2011

2 - Energy Efficiency Plan - COM(2011) 109

3 - Communication from the Commission COM (2007) 723 final, see this link: http://eur-lex.europa.eu/LexUriServ/LexUriServ.do?uri=COM:2007:0723:FIN:EN:PDF

4 - Communication from the European Commission (COM 2011) 885 final, see this link: http://ec.europa.eu/energy/publications/doc/2012_energy_roadmap_2050_en.pdf

5 - European Nuclear Society – July 2012: http://www.euronuclear.org/info/encyclopedia/n/nuclear-power-plant-europe.htm

Context of energy policy

Europe has to face three energy challenges: security of energy supply, limitation of greenhouse gas emissions and competitiveness of energy-reliant economies, while keeping the global temperature increase below 2ºC and thus avoiding dangerous impact on climate.

The European Council committed in March 2007 to very ambitious goals putting Europe at the forefront of the fight against climate change. The "20-20-20" objectives for 2020:

- 20% reduction in greenhouse gas emissions compared to 1990
- 20% energy savings
- 20% share of renewable energies in the total energy mix

The EU is currently on track to meet two of those targets, but will not meet its energy efficiency target unless further efforts are made[2]. The European Council has also given a long-term commitment to the decarbonisation path with a target for the EU and other industrialised countries of an 80 to 95% cut in emissions by 2050 compared to 1990 levels.

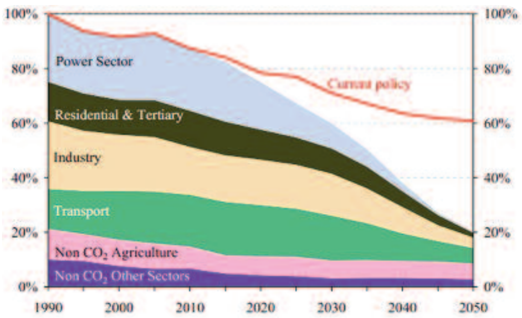

Figure 7: EU GHG emissions towards an 80% domestic reduction (100% = 1990) (Source: Low-Carbon Economy Roadmap March 2011 - European Commission)

To achieve both the medium term goals (2020) and the long term vision (2050), Europe launched in 2007 the "Strategic Energy Technology (SET) Plan" which identifies a list of competitive low carbon energy technologies to be developed and deployed in Europe[3].

The SET plan identifies nuclear fission as one of the key low carbon energy technologies which Europe must develop and deploy. For the 2020 objectives, the intention is to "maintain the competitiveness in fission technologies together with long term waste management solutions". This can be translated as maintaining at least the current level of nuclear energy in Europe's electricity mix (around 30%) through long-term operation of existing plants and an ambitious programme of new build.

For the vision of 2050, the SET Plan recommends us to act now to "complete the preparation for the demonstration of a new generation of fission reactors with increased sustainability".

Role of nuclear energy in Europe

The Communication from the European Commission in December 2011 entitled "Energy Roadmap 2050"[4], recognises the importance of nuclear energy's contribution in Europe today. With approx. 122 GWe in operation in Europe, that equates to 30% of electricity generation (produced by more than 131 reactors located in 14 countries in the EU-27[5]), nuclear fission represents the largest low-carbon energy source in Europe (2/3 of the decarbonised electricity).

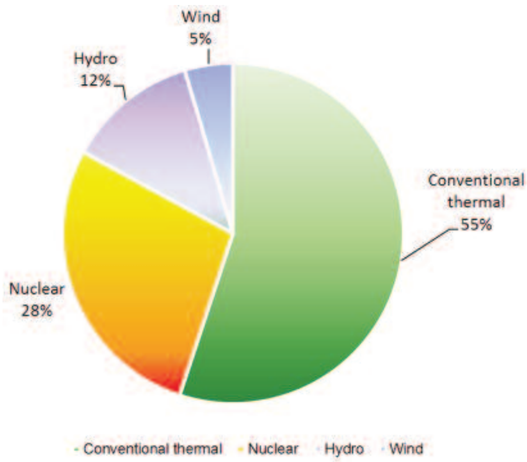

Figure 8: Electricity generation shares in EU-27 in 2011 (Source: Eurostat / Electricity production and supply statistics)

Thus, nuclear energy is the most important low carbon technology in Europe's energy mix. It is estimated (see the platform's Vision Report) that compared to a representative mix of alternative base-load capacity (essentially gas and coal), Europe's nuclear power plants represent a saving of almost 900 million tonnes of CO_2 per year, i.e. approximately the level of emissions from the whole transport sector.

Nuclear energy generates around one-third of EU electricity and two-thirds of its carbon free electricity

Access to secure and affordable energy is vital if society is to meet these needs. Because of its polyvalence, access to electricity is particularly important.

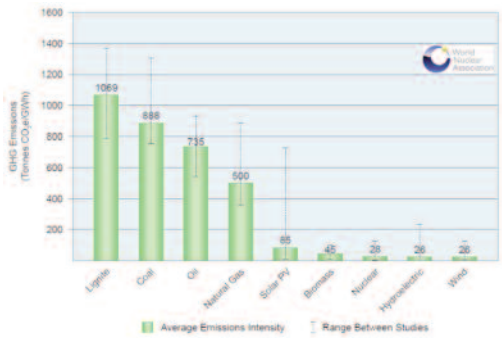

Figure 9: Greenhouse gas emissions (in tonnes of CO_2-equivalent) per GWh for different electricity production means, (Source: World Nuclear Association)

To be sustainable, energy production must avoid endangering the well-being of future generations, not only by reducing the use of natural resources but also by minimising detrimental effects on public health and the environment, including the production of ultimate waste. In particular, electricity production must achieve high levels of safety and limit harmful emissions over the full lifecycle of the plant (cradle to grave).

Nuclear energy is the most cost competitive carbon free form of electricity generation. The share of new power generation and investment shows how renewables are often capital-intensive, representing 60% of investment for 30% of additional generation.

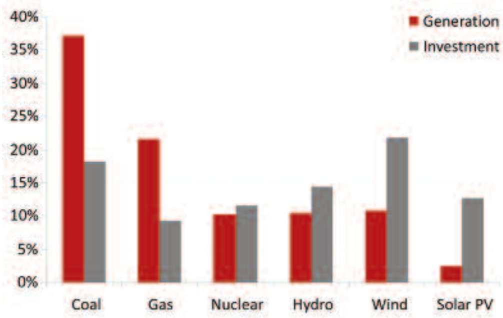

Figure 10: Share of new power generation & investment in 2011. (Source: World Energy Outlook, IEA)

Furthermore the European nuclear industry is a big job creator in the low carbon energy mix. Based on the scenario "Delayed CCS" of the EU Energy Roadmap 2050 (where nuclear contributes nearly 20% in 2050) the nuclear industry will create 347,000 additional jobs in Europe coming from Lifetime extension, new build, decommissioning and geological disposal programmes, over and above the jobs created by the regular operation (900,000 jobs). The corresponding total "valued added" for the European economy can be estimated to €70 billion per year[6].

Security of supply is a key factor in the role of nuclear energy. With the current efficiency of uranium in nuclear power plants and at the projected 2012 rate of consumption, the natural resources may last approximately 100 years[7], depending on the nuclear power growth rate in the coming decades. The security of supply will be assured for thousands of years if fast neutron reactors with closed fuel cycle are deployed.

The Fukushima accident increased public concern about nuclear energy and drew a new attention to the safety of nuclear power plants.

As an immediate action after the accident, the SNETP Governing Board decided to set up a dedicated task force to assess the lessons learned from the Fukushima accident, in order to identify the appropriate measures and adaptations of SNETP's work programme. The results and conclusions of the Fukushima Task Force report are reflected in this edition of the SRIA.

SNETP considers nuclear safety as its prime priority and responsibility in front of the European citizens. Nuclear energy is and will need to remain a key element in meeting Europe's needs for security of energy supply, competitiveness, and the fight against climate change.

Mandated by the European Commission, the European Committee for Standardisation - CEN - and the European Committee for Electro-technical Standardisation - CENELEC - are entitled to develop European Standards (EN) for the nuclear sector. They encourage European research and development activities to feed relevant results into the international standardisation organisations and support new initiatives to enlarge the process of standardisation of nuclear codes.

Technology Platforms

European Technology Platforms were established by the European Commission as industry-led stakeholder forums in 2003 to promote research and development in technological domains. They *"provide a framework for stakeholders, led by*

Technology platforms play a key role in ensuring the high impact of EU research in leading markets and technological areas

6 - Socio economic role of nuclear 2020-2050 http://ec.europa.eu/energy/nuclear/forum/opportunities/doc/opportunities/2012_04_04/socioeconomic_role_nuclear_2020_2050_final.pdf

7 - Uranium: Resources, Production and Demand "Red Book" 2011, OECD's NEA and IAEA

industry, to define research and development priorities, timeframes and action plans on a number of strategically important issues where achieving Europe's future growth, competitiveness and sustainability objectives is dependent upon major research and technological advances in the medium to long term".[8]

Technology Platforms play a key role in ensuring the high impact of EU research in leading markets and technological areas with the overall objective of closing the gap with global innovation leaders and driving jobs and growth. European Technology Platforms will support the European Commission in implementing The Innovation Union and Horizon 2020.

8 - http://cordis.europa.eu/technology-platforms/home_en.html

The Strategic Research and Innovation Agenda is one of the most important outputs from a Technology Platform, as it provides decision-makers as well as the scientific community at large with research, development and demonstration roadmaps to achieve a shared vision. Due to its vital role, it is important to keep the document up to date.

SNETP 2007 - 2012

The Sustainable Nuclear Technology Platform is now nearly six years old. During this time, SNETP has achieved efficient collaboration between its stakeholders and has also developed a common vision regarding the future contribution of nuclear fission energy in Europe. The three pillars of the platform have evolved in different ways:

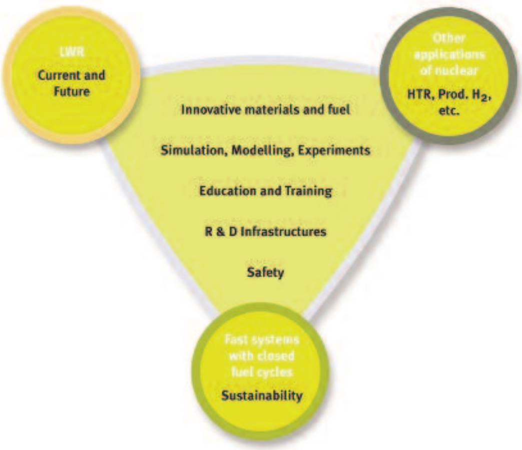

Figure 11: The SNETP three pillars

- NUGENIA is now, since its launching in March 2012, an international association mandated by SNETP. Its main role is to help develop R&D supporting safe, reliable, and competitive second and third generation nuclear systems. NUGENIA has currently 59 members from 17 countries.
- The European Sustainable Nuclear Industrial Initiative (ESNII) was officially launched in November 2010 under the SET Plan. ESNII promotes advanced reactors with the objective of resource preservation and the minimisation of the burden of radioactive waste. The Initiative has already 22 members and is continuously growing with its four projects: ASTRID, MYRRHA, ALFRED and ALLEGRO.
- The Nuclear Cogeneration Industrial Initiative (NC2I) aims at demonstrating an innovative and competitive energy solution for the low-carbon cogeneration of process heat and electricity based on nuclear energy. NC2I was officially launched at the SET-Plan Conference in November 2011. Recently, a group of industrial heat users manifested its interest in High Temperature Reactor (HTR) technology, although other technologies are not excluded. International cooperation (e.g. with China or the US) could lead to a joint development of a demonstrator.

Public and stakeholder communication

SNETP also identifies efforts to resolve the concerns raised by the public and stakeholders.

Nuclear safety is the first priority of all activities, being a mandatory condition and an underlying objective.

The nuclear industry makes more and more efforts to inform populations around nuclear installations, in particular through the use of local information committees.

Against the background of the Aarhus Convention, the national authorities of all European Member States engaged to provide information to any citizen representative body, and to listen to the questions raised regarding the potential risks of nuclear installations.

This may involve developing new research items in the field of social sciences, but also highlighting some new technical fields for studies devoted to answering questions from stakeholders.

Structure of the Strategic Research and Innovation Agenda

This new Strategic Research and Innovation Agenda of SNETP maintains the objective of its first edition to address the short-(around 2015), medium-(around 2020) and long-term challenges (2050), as the SET-Plan does, with respect to fission technologies.

The first chapter of the updated version of the SRIA is focused on safety. The Fukushima accident has raised public concern about nuclear energy and has drawn more attention to the safety of nuclear power plants. This chapter identifies the challenges nuclear energy is currently facing in this new situation and the relevant areas of R&D that will play an essential role in future utilisation of nuclear power.

The second chapter addresses the R&D challenges for improving the current fuel cycles. The current reactors are only able to use less than 1% of the uranium extracted in nature. To enhance the use of uranium and minimisation of the final waste the closed fuel cycle and advanced neutron reactors play a vital role. This chapter identifies those R&D measures that optimise natural resources in the short term.

The third chapter deals with the R&D to make advances in the safe, reliable and efficient operation of nuclear power plants, which is covered by NUGENIA. Key issues have been identified to meet safety requirements. This chapter is structured in line with NUGENIA's eight Technical Areas: plant safety and risk assessment; severe accidents; improved reactor operation; Integrity assessment of systems, structures and components; fuel, waste management and dismantling; innovative LWR Generation III design; harmonisation; and in-service inspection and inspection qualification. This chapter also focuses on the current state of the art and the challenges nuclear energy has to face in the short and medium term.

ESNII fast reactor systems are treated in the fourth chapter. The European Sustainable Nuclear Industrial Initiative focuses on the more sustainable development of nuclear energy and its fuel cycle. This chapter deals with the current status, state of the art and R&D challenges of the four ESNII projects: ASTRID, the Sodium Fast Reactor Prototype, MYRRHA, a flexible fast spectrum irradiation facility, ALFRED, the Lead Fast Reactor Prototype and ALLEGRO, the Gas Fast Reactor Prototype.

The fifth chapter addresses another potential market for nuclear energy, the cogeneration of heat and electricity using a nuclear reactor, mainly for industrial heat application. Fossil fuels are currently the main source of heat for European industry. This chapter addresses High Temperature Reactor (HTR) technology, due to its large potential for these new applications, as well as other technologies.

The sixth chapter focuses on cross-cutting activities: fuel reprocessing and Education and Training. Technical topics addressed as cross-cutting in the SRA 2009, like materials, pre-normative research, simulation tools and infrastructures have been integrated in the three main technological chapters.

This new edition of the SRIA is compatible and complementary to the SNETP vision report, the original SRA, and associated documents, but more detailed descriptions are provided for the R&D needs to be performed in the short and medium term. Longer-term topics, described in less detail in this update are not less important. Indeed, the SNETP's Strategic Research and Innovation Agenda is a living document and will continue to be periodically reviewed and updated.

Safety vision

1. Introduction

Dedicated research and development, and pervasive safety culture, are key factors in the permanent process of improvement of nuclear installations and their safety.

Safety of nuclear installations has undergone continuous efforts since the beginning of the nuclear era and has been since then a relevant driving force for research and development nuclear technology. This trend is expected to be strengthened in the future also as a consequence of the recent events at Fukushima.

Needs for safety research are expressed by the main stakeholders - designers, operators, regulators and TSOs - from their respective perspective. Part of this research - mainly addressing phenomenology, data-base generation and modelling - can be performed jointly, also relying upon contributions from universities and research centres, to allow the substantiation of common knowledge, while the complement - addressing design, operation and applied technology issues - is to be carried-out independently, to guarantee full independency among the actors in the nuclear field.

Dedicated research and development, and pervasive safety culture, are key factors in the permanent process of improvement of nuclear installations and their safety

Up to now, the bulk of safety research has been carried out within national programmes supported by either public financing schemes, or by the operators, in the respective fields of endeavour. Presently, the R&D effort is largely shared through EURATOM framework programmes and, at international level, through the OECD/NEA programmes, and in compliance with the IAEA programmes and guidelines.

The safety related R&D has been already addressed as a cross-cutting issue, in the 2009 edition of the SNETP Strategic Research Agenda, which identified a series of safety-specific tasks both for current and advanced reactor systems.

Thanks to the work performed within SNETP - the Gen II/III WG, the NUGENIA association and the ESNII and Cogeneration task-groups the present release of the SNETP Strategic Research and Innovation Agenda is able to present a more integrated selection of the issues that Platform members acknowledge as ranking at highest priority, and also suitable for harmonisation at European level.

Looking ahead, and considering in particular the outcomes of the Fukushima accident, the regulators - according to the WENRA objectives[9] - are to require enhanced safety for reactors already in operation as well as for those either in construction or to be built. On the other hand, the operators will continue to be seeking continued amelioration of the efficiency and availability of the plants e.g. through improved fuel utilisation, nominal power increase and extended service time.

To address these new and more demanding requests, pre-normative research is to be promoted within SNETP to allow achieving the inherent safety objectives, thus providing the stakeholders with better optimised definitions of safety criteria. Moreover, suitable R&D programmes are needed to support any possible evolution of operational practices.

More generally, research should contribute to allowing best estimate evaluation of reactor system behaviour beyond design loads, up to the cliff edge threshold resulting in their complete degradation.

The present chapter takes into account the lessons already learnt from the Fukushima

9 - http://www.wenra.org/media/filer_public/2012/11/05/wenra_statementonsafetyobjectivesfornewnuclearpowerplants_nov2010.pdf

accident so as to identify the need for new R&D programmes and/or new aspects to complement the programmes already ongoing, also accounting for the results of the work already carried out since publication of the first SRA in 2009.

2. The Fukushima accident and its implications

The accident occurring at the Fukushima Daiichi nuclear power plant on 11 March, 2011 has raised public concern about nuclear energy and has drawn new attention to the safety of nuclear power plants, in particular in the case of extremely severe external hazards.

The accident was triggered by a combination of two main initiating events:

- An exceptional magnitude earthquake which caused the sudden loss of almost all the off-site power supply. The reactors 1-2-3 were automatically shut down. The residual heat removal systems were started immediately, relying on electricity supplied by emergency power sources (diesel generators and batteries).
- The associated tsunami caused the flooding of the site under a wave about twice the size considered previously in the risk evaluation, highlighting a serious underestimation of the risk by the operator as well as a defect in the supervisory function of the safety authority. The wave led to both the loss of all the emergency power supply systems and of the heat sink.

The immediate challenge for the emergency response team was to recover cooling capabilities in a situation where the off-site power supply required about 11 days to be effective again. All the reactors in operation or loaded with fuel were affected and experienced core melting of various degrees, hydrogen explosions, radioactivity release and contamination of air, ground and seawater. The spent fuel pools were affected as well.

A wide activity to reconstruct the accident scenario, investigate the phenomenology of the events and pile-up data has already been carried-out and is presently ongoing. It will continue for several years, on the basis of new information which will be made available, and it will contribute to enhancing knowledge and improving the reliability of computation through validation, especially for severe accident simulation tools. It will also provide a valuable input to the upcoming improvement of severe accident management and emergency preparedness measures.

The defueling, decommissioning and decontamination activities on the Fukushima Daiichi site will take many years, likely 30 to 40, to be completed. They will also require many actions, some of which have already been identified by TEPCO, mainly the development of:

- inspection methods and devices for inspection of leakage points in the containment vessel, and for operation in high-temperature, high humidity and high-dose environments
- remote sampling and decontamination methods
- technologies and methods to repair leakage points, including underwater repair
- systems to prevent the dispersal of radioactive materials
- core defueling and fuel debris removal technologies

These needs will likely represent an opportunity for R&D cooperation between Europe and Japan and for the integration of this knowledge into European organisations.

Even if a fully detailed analysis of the system deficiencies is still difficult, there are already lessons learned providing indications for future R&D and safety improvements. The investigation of the Fukushima accident is in fact generating a new scale of priorities with specific focus on managing extreme external events and their combinations, on common mode failures and human behaviour, and on the assessment of their impact on the robustness of defence in depth. Results from this R&D can then be translated into real safety upgrades.

3. Outcomes of the European Stress Tests

A number of initiatives have been undertaken in many countries and at international level to take into account the lessons learned from this accident for the improvement of nuclear reactor design and safety provisions and for the improvement of the organisation for managing an accidental situation.

Most of the countries operating nuclear reactors have launched systematic reassessments of the safety margins of their nuclear fleet under severe natural hazards. The European Council of 24-25 March 2011 requested a comprehensive safety reassessment to be performed on all EU nuclear power plants, with respect to extreme initiating events and consequential loss of safety functions. The request of the Council included safety reassessments performed at national level, complemented by a European peer review.

This multilateral exercise covered over 150 reactors in European countries operating nuclear power plants. The stress tests focused on three topics:

1) natural initiating events, including earthquake, flooding and extreme weather
2) loss of electrical power and loss of ultimate heat sink
3) severe accident management

The stress tests consisted of three steps. In the first step, following the ENSREG specifications, the plant operators performed an assessment and made proposals for safety improvements. In the second step the national regulators performed an independent review of the operators' assessments and issued additional requirements. The last step was a European peer review of the national reports, consisting of a desktop review, followed by a two week assessment of the report during topical review, completed by additional discussions during the country reviews.

As a result of the stress tests, 17 national reports by the respective safety authorities, a peer review report for each of the seventeen participating countries, and a final peer review report (developed by the Stress Test Peer Review Board and endorsed by ENSREG on 26 April 2012[10]) were prepared. The review focused on the identification of strong features, weaknesses and possible ways to increase plant robustness in the light of the preliminary lessons learned from Fukushima.

Although the stress tests were based on information available at the time and were not primarily intended to specify areas for future research, they nevertheless indicated the need for future studies and developments.

4. Priority research items identified by the SNETP Fukushima Task Group

In the framework of this enhanced global effort, the SNETP Governing Board empowered a Task Group to investigate how the lessons learned from the Fukushima accident could impact safety related R&D orientations and priorities. The Task Group concentrated on short and medium term R&D, in particular on the development, updating and validation of methods and tools for areas which are not considered as enough understood or covered, and has issued a report[11] which, even if it can be considered as a first step of the process, already catches very important features which are summarised here below.

According to the conclusions of the Task Group, no really new phenomena were revealed by the Fukushima accident

On a longer timescale – several years – the outcomes of future expertise acquired from the reactors and fuel pools of Fukushima Daiichi site will be very valuable for the qualification and validation of the results of the R&D tasks and for the definition of new targets.

According to the conclusions of the Task Group, no really new phenomena were revealed by the Fukushima accident. However the TG identified 13 main research subjects to be considered with the appropriate priority, in the following areas:

a) plant design and identification of external hazards
b) analysis and management of severe accidents
c) emergency management and radiological impact

In particular, the issues related to extremely severe and rare accidents will to be considered in a more global approach to safety in order to better understand the design margins and the behaviour of nuclear reactors under beyond design basis scenarios.

The 13 items are:

a) systematic assessment of vulnerabilities to defence-in-depth and safety margins for beyond design basis loads
b) human and organisational factors under high stress and harmful conditions

10 - Joint statement of ENSREG and the European Commission on 26 April 2012

11 - "Identification of Research Areas in Response to the Fukushima Accident" 2013, Fukushima Task Force

c) improved methods for external event hazard evaluation
d) use of the probabilistic methods to assess plant safety in relation to extreme events
e) advanced deterministic methods to assess plant safety in relation to extreme events
f) advanced safety systems
g) material behaviour during severe accident
h) advanced methods for the analysis of severe accidents
i) improved procedures for management of severe accidents
j) assessment of the radiological effects of severe accidents
k) improved modelling of fuel degradation in spent fuel pools
l) methods for minimisation of contamination in the NPP surroundings and for treatment of large volumes of radioactive waste
m) accident management in the framework of the integrated rescue system

Special attention shall be devoted to how the research outcomes will be implemented and transferred into standard industrial practice.

5. New directions and challenges in safety research

More inputs from the analysis of the Fukushima event will be generated in the coming years both from the analysis of the situation on the site and from the application of simulation codes to the accident. It can be however already affirmed that the main challenges identified from the lessons learned from the accident are the following:

- To better characterise any natural events, like earthquakes, floods, etc., including methodologies for dealing with rare events.
- To extend even more in-depth the safety approach to any type of initiating event, especially severe natural hazards and any combination of them. It shall be done for current reactors, Generation III reactors and future reactors.
- To include beyond design basis accidents more systematically at the design stage to assure the robustness of defence in-depth and to avoid cliff edge effects. The approach shall include situations where all units on the same site are affected by a beyond design event.
- To develop wider and more robust lines of defence with respect to design basis aggressions and beyond design basis events to define additional measures to consider in the design and to improve or to develop systems for mitigation of consequences.

It is therefore expected that safety analysis of the nuclear plants will evolve in the future, defining more advanced objectives, like

- evaluation of the best estimate behaviour of the plant systems during beyond design basis accidents, to assess possible challenges to the fulfilment of safety functions
- evaluation of the ultimate capacity of the systems with respect to the load applied and to identify when the level of damage becomes non-linear or catastrophic

This will require an extension of the capability of physical modelling and computer tools in different areas and in particular in the area of severe accidents and containment system simulation enhanced in order to derive the radioactive source term due to the accident. Both experimental and theoretical research efforts will be necessary to support the possible evolution of safety regulations and practices.

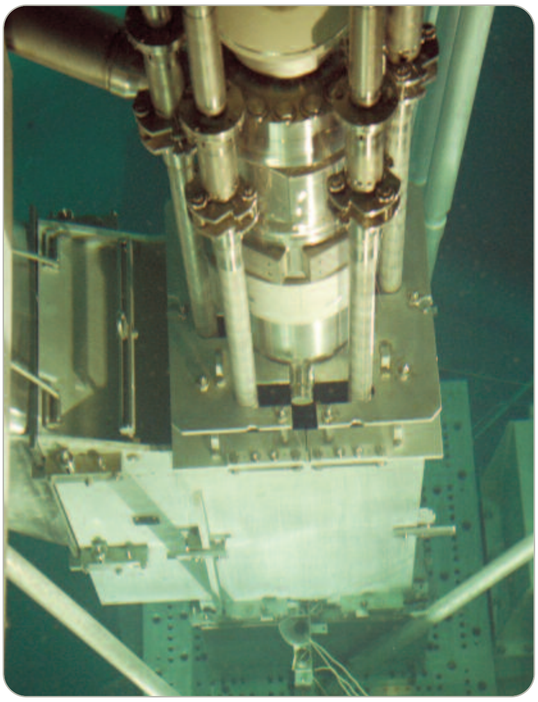

Figure 12: PHEBUS (Source: IRSN)

Specific emphasis will have to be put also on emergency management, which has been challenged during the Fukushima accident due to:

- the concomitance of many events, the severe environmental conditions and the mutual interaction between the affected units on site

- the complexity and difficulty of the decision making process which altered the effectiveness and promptness of actions and which generated both confusion and delays
- the practical impossibility of recovering an adequate and stable electrical supply source for several days

The improvement of emergency preparedness and response shall include the consideration of several items:

- the availability of more sophisticated tools to provide the operators with more reliable and quicker indications/measurements on the reactor status to help in the implementation of an appropriate recovery strategy
- the availability of redundant intervention means in the vicinity of the site
- the availability of better and faster environmental monitoring systems, better models for predicting contamination, health effects of low doses, and the effect of contamination on the environment – in particular the marine environment
- a broader consideration of social issues and cultural aspects
- better international cooperation/expertise which could provide help on plant status diagnostics, on the forecasting of accident evolution and on mitigation strategies

The 13 items and new directions and challenges identified by the Fukishima TF are discussed in the R&D needs and priorities within the following chapters.

Sustainability of the nuclear Fuel Cycle

1. Introduction

A large number of studies have been carried out worldwide, and particularly in Europe but also within the framework of the Generation IV roadmap, to analyse the meaning of "sustainability"[12] when applied to the NFC. From that work, there is clear consensus today that a sustainable NFC is mainly linked to the durability of the solutions addressing the three following issues:

- optimum use of natural resources
- nuclear waste minimisation
- minimum impact on environment

These objectives must be pursued while at the same time maintaining or increasing safety, economic competitiveness and protection against diversion or undeclared use of nuclear material or technology.

Present light water reactors are only able to use less than 1 % of the mined uranium. With such a low efficiency, the presently identified worldwide U resources are sufficient for about 100 years[7], depending inter alia on the acceptable uranium cost and on the nuclear power growth rate during the coming decades. In order to make nuclear fission energy sustainable in the long term, new technological solutions improving the usage of this natural resource by around 50 times are being developed.

The new technology for waste minimisation and resource optimisation is based on the combination of fast neutron systems with multi-recycling of the fuel in advanced fuel cycles. This will be achieved while maintaining or improving safety and economic competitiveness, and minimising risks of proliferation.

The new technology for waste minimisation and resource optimisation is based on the combination of fast neutron systems with multi-recycling of the fuel in advanced fuel cycles

The change to enhanced sustainability will be, most probably, a progressive process that has already started. As a matter of fact, some of the technologies for recycling fuels are commercially available and industrially operative in particular in some EU countries, e.g. France, UK. The combination of these and other existing technologies with improvements in the present reactor designs allows to progress towards both optimised use of natural resources and economic competitiveness.

This chapter describes the potential and required R&D associated with these advanced fuel cycles.

12 - The general UN definition of sustainability is « Meeting the needs of the present without compromising the ability of future generations to meet their own needs"

2. Nuclear Fuel Cycle

In the broadest sense, the Nuclear Fuel Cycle (NFC) encompasses all steps and facilities needed to produce electricity in nuclear reactors, including uranium mining and preparation of the fuel which will be used in nuclear reactors, that is the "front end" of the fuel cycle, and the "back end" or management of the fuel after its use in the reactors (the spent fuel), with two main options (both implemented in Europe):

- direct disposal of spent fuel, called the "open cycle"
- recycling of valuable materials, called the "closed cycle"

More precisely, the nuclear fuel cycle includes the following steps:

- The "front end" of the fuel cycle, which consists of uranium (or thorium) prospection, mining and on-site purification, uranium (or thorium) conversion (to obtain pure UF_6, UO_2 or metal, depending on its future use), uranium enrichment (if needed), and fuel fabrication.

- Fuel irradiation in nuclear reactors to produce electricity and heat.
- "The back end" of the fuel cycle, which consists of interim storage of spent fuels, recycling (which includes reprocessing of the spent fuel to recover recyclable materials and fabrication of new fuels with these materials, if this option is implemented), transportation of radioactive materials (spent fuels, conditioned radioactive waste, etc), final disposal of nuclear waste (spent fuel for "open cycle" option or ultimate waste for the "closed cycle").

3. R&D to improve sustainability of Nuclear Fuel Cycles

3.1 Optimum use of natural resources: from the short to the long term

There are only two kinds of "natural resources" for nuclear fission:

- Uranium, containing mainly 2 isotopes: one fissile, ^{235}U, which constitutes only 0.71% of natural uranium and one fertile, ^{238}U, which constitutes 99.29 % of natural uranium.
- Thorium, containing exclusively one isotope, ^{232}Th, which is a fertile isotope (producing the fissile isotope ^{233}U). Although technically possible, the fuel cycle based on thorium requires an initial supply of a fissile isotope (^{235}U or plutonium) to be deployed and is not implemented on an industrial scale today in any European country.

The optimisation of natural resources is progressively achieved by the industry at each step of the Nuclear Fuel Cycle

The optimisation of natural resources, to maximise the electricity obtained per unit of uranium mined, is progressively achieved by the industry at each step of the NFC, through operation of the market and with growing technical knowledge. This is the case for example in the front end through the selection of cut grade of uranium deposits or tail enrichment; by fuel management inside the reactor; by improving reactor designs; or by spent fuel recycling, in the back end.

Optimisation at each step of the NFC implies R&D programmes. However, the "front end" steps of the NFC, such as uranium prospecting and mining or enrichment process and fuel fabrication (UO_2), are more a matter for industry, and in the phase of commercial competition. Consequently, the SRIA will rather focus on enhancing the usage of mined uranium and generated plutonium in present and future reactors, and on the NFC back end options.

An efficient way of reducing the natural uranium consumption is to increase the reactor conversion ratio and to recycle fissile material

Nuclear reactors are able to convert a part of the fertile isotopes which are loaded in the fresh fuel into fissile isotopes, e.g. ^{238}U into ^{239}Pu. The ratio between the total amount of artificial fissile material created inside the reactor core and the total amount of fissile isotopes "consumed" is called "conversion ratio". A part of the artificial fissile isotopes is burned in situ contributing to the generation of electricity and saving natural fissile isotopes. However, the part of the created artificial fissile isotopes which is not burned in situ remains in the spent fuel when unloaded. The recycling of this part can further contribute to saving natural fissile isotopes.

Consequently, one of the most efficient routes to reducing natural uranium consumption is to increase the conversion ratio of present and future reactors and to recycle fissile material.

First improvements in the use of natural resources could be made using advanced light water reactor systems with higher conversion ratio and improved fuel design, and the associated back end of the NFC, as well. Attractive opportunities have already been identified for Advanced Generation III reactors, such as new core designs and loading strategies, fuels accommodating very high burn-ups (>70 GWd/tHM) for LWR and improved capacity for plutonium and reprocessed uranium recycling, including 100 % MOX cores, plutonium multi-recycling for LWR and 100% plutonium cores for HTR. The corresponding R&D challenges, included in the objectives of the recently created NUGENIA association, are discussed in the corresponding chapter.

However, fast nuclear reactors can be designed to reach conversion ratios equal or even greater than one, in such a way that no more natural fissile isotope is needed to sustain nuclear energy, since the reactors generate more fissile isotopes than they consume to produce energy. These reactors, called "breeders" need to be fed at equilibrium with fertile isotopes (^{238}U or ^{232}Th) which are available in plentiful amounts, both in nature and as leftovers from the present enrichment of nuclear fuel. In this way, fast reactors will allow increasing the use of the natural uranium resources by a factor of around 50.

Therefore it must be underlined that "breeder" reactors, in practice Fast Neutron Reactors (FNR)[13], are the only solution which can lead to the long term sustainable development of nuclear energy, with regard to the "optimum use of natural uranium resources". Because of this critical role in nuclear energy sustainability, a specific chapter of this SRIA is devoted to developments on FNRs, where the associated R&D for feasibility and natural resource optimisation are addressed.

Nuclear waste consists of radioactive residues produced by electricity generation in fission reactors and are considered as not reusable

The efficiency of those technologies depends on the availability of other facilities of the fuel cycle like those for reprocessing discussed later in this chapter, and of deployment strategies. The efficiency of different deployment strategies and the appropriate coupling of the different elements of the nuclear fuel cycle should be optimised with the help of scenario studies.

3.2 Nuclear waste minimisation

Nuclear waste (NW) is classified in various ways in different European countries, but generally, according to its intrinsic risk and management route. The main parameters[14] are the level of specific radioactivity, the decay "half life" and the specific heat produced by the radioactivity of the unstable isotopes contained in the NW. Industrial solutions to handle Low Level Waste (LLW) and most of the Intermediate Level waste (ILW) are already implemented in several EU countries.

High level waste (HLW), which contains highly radioactive isotopes, significant quantities of long lived radio-nuclides, and is strongly heat emitting, is mainly generated by the operation of nuclear reactors. HLW can be the spent fuel, or waste from its reprocessing or from other steps of the NFC. The present solution for HLW is to properly condition it inside isolating and protecting packages that are then disposed of in a Deep Underground Geological Repository (DGR). A number of technological and geological barriers are set up in this way to avoid any hazard to the population or the biosphere. This solution has been scientifically proven to be reliable and safe, and most of its technologies are ready for deployment. The first implementations in the EU are expected in Finland, Sweden and France within the next 7-20 years. The associated R&D, technology development and implementation are the topics of another Technology Platform (Implementing Geological Disposal, IGD-TP) and will not be further discussed here.

To optimise HLW management, research should focus on minimising several parameters of the HLW:

- the mass and volume of conditioned NW to be disposed of
- the long term "radiotoxic inventory", which is the sum of activities of each radioisotope in the NW weighted by the dose factor that indicate the risk if this material would be dispersed within the population
- the effective "lifetime" of conditioned NW
- the heat generation of conditioned NW as function of time, due to the radioactivity of its unstable radioisotopes. This parameter strongly affects the DGR capacity
- the "long-term radiological impact", that is the calculated biological effect on living species of possible radioactive releases into the biosphere once part of radionuclides (or their radioactive daughters) has reached the surface

A first way to minimise nuclear waste is to reduce the amount of radio-nuclides produced by nuclear reactors. For fission products the production is directly proportional to the electricity generation, so that the only way to reduce their amounts is to increase the electrical efficiency of nuclear power reactors. Cogeneration with direct utilisation of the nuclear heat, e.g. for industrial process, could be used to reduce the amount of NW per unit of useful energy generated. On the other hand, there are several means to influence the production of the different actinides, including

13 - In principle breeder could also be designed with thermal neutron energy spectrum using thorium based fuels.

14 - The specific radioactivity of one object is the number of disintegrations per unit of time in a given unit volume or in a given unit of mass. The half-life on an isotope is the time interval required for its radioactivity to get reduced by half.

the choice of reactor type (neutron spectrum) or even the choice of fuel cycle (for example, thorium based fuel which could generate much smaller amounts of minor actinides in the long term).

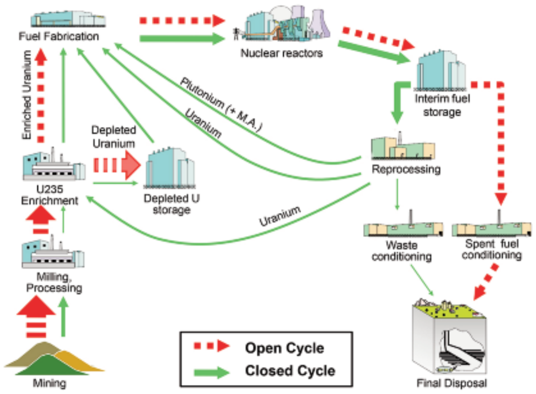

Figure 13: The closed nuclear fuel cycle illustrating the recycling and optimisation of energy resources and the minimisation of nuclear waste (Source: CIEMAT)

Once the waste has been produced, if the spent fuel is directly disposed of, there is in fact no way to act on the previously indicated optimisation parameters, except the enhancement of confinement and its durability (waste matrix or waste container) and the time before disposing of the NW. This research could also contribute to reducing the “long-term radiological impact”. These topics fall outside the scope of the SNETP, but are addressed by IGD-TP SRA in its key topic 2 "Waste forms and their behaviour".

If the spent fuel is reprocessed, many technical options are open to produce improvements in the five NW parameters quoted above.

The general conclusion is that waste minimisation in advanced fuel cycles should be considered within a global objective of sustainability

In this regard, studies, carried out, in particular, under European R&D programmes, such as RED-IMPACT and PATEROS, and the conclusions of the NEA/OECD expert groups from the Working Party on scientific issues of Advanced Fuel Cycle (WPFC), have shown that one of the most promising routes is the “Partitioning and Transmutation” of selected radio-nuclides (particularly actinides). The general conclusion is that waste minimisation in advanced fuel cycles should be considered within a global objective of sustainability. Furthermore the implications on the reduction of the number and size of DGRs and other societal aspects need to be considered in the fuel cycle optimisation.

In this sense, three types of objectives are identified:

- integral management of all actinides in a long term sustainable nuclear fleet
- integral reduction of the transuranic actinide inventories
- specific reduction of some Minor Actinide inventories (Np, Am and possibly Cm)

These objectives can be achieved conceptually in two generic types of scenarios:

- A fleet of fast neutron spectrum critical reactors that simultaneously produce electricity and transmute all the actinides. The only input into the system (reactors and fuel cycle facilities) is natural or depleted uranium and the output is electricity and residual HLW plus ILW, including the fission fragments, activation products and actinide reprocessing losses. In this option, the MA could be homogeneously diluted within the whole fuel or separated in the form of dedicated targets. However the core design of these reactors has to be optimised from the point of view of neutron economy and safety performance and, in addition, the feasibility of the associated fuel cycles should also be addressed.
- A “double strata” reactor fleet. The first stratum consists of a set of critical reactors dedicated to electricity production using “clean fuel” containing only U and Pu. The reactors in this stratum can be either present or future thermal reactors or fast reactors or an appropriate combination of both. The second stratum, devoted to TRU or MA transmutation and representing a small fraction of the total installed power, would be based on special low conversion ratio fast reactors or subcritical fast systems, ADS. These reactors would be loaded with homogeneous fuels with high MA content, and would have to be optimised from the point of view of neutron economy and safety performance.

The process of deployment of these advanced fuel cycles with partitioning and transmutation will necessarily be progressive. In first instance, economic competitiveness will favour the life extension of present reactors or their replacement with advanced Generation III LWRs. Later, as U resources become scarcer and waste inventories grow, the fast nuclear systems (FNR and/or ADS) will appear more attractive and will eventually be progressively introduced.

The evaluation of this type of scenario indicates that while maintaining the safety of operation, they should ultimately be able to strongly reduce long term uranium consumption, making the present reserves last for several thousand years. At the same time, the long term radiotoxic inventory of HLW could be reduced by more than a factor of 100 and its heat load by more than a factor of 10. According to available studies, quoted above in this section, the last figure will allow the DGR size to be reduced by factors from 3 to more than 10 (in hard rock, clay and tuff geological formations). In the case of large nuclear reactor parks, waste minimisation could help to minimise the number of required DGRs. Countries with smaller fleets might need to participate in regional solutions involving cooperation with a country with a large nuclear fleet to improve the partitioning and transmutation efficiency and its economic feasibility.

The deployment of these advanced fuel cycles involves large technological challenges on:

- new fuels (targets) and fuel assembly designs bearing significant amounts of MA, and their fabrication technology
- the technologies of FNR and ADS, including new materials, thermal-hydraulics, simulation tools, nuclear data and, in the case of ADS, the coupling of an accelerator with a subcritical core
- new recycling technologies based on advanced aqueous and pyro-metallurgic reprocessing technologies, adapted to highly active and hot fuels containing large amounts of Pu and MA, and minimising the production of secondary wastes

The first two points are further developed in the ESNII chapter for each of the fast system types. On the other hand the development of the reprocessing of irradiated fuel is discussed in the chapter dedicated to cross cutting activities.

3.3 Advanced fuel cycle scenario research

A continuous effort on Scenario Studies of nuclear material management and the impact of advanced fuel cycle technologies on the final Deep Geological Repository, DGR, is maintained by several laboratories in different EU Member States, and as contributions to NEA/OECD and IAEA studies. Regional and global considerations as well as transition effects are important aspects of all studies recently performed by SNETP members or which are under discussion.

However, additional efforts are required to complete studies on the feasibility of sustainable solutions for the transition period from the present nuclear fleet until the deployment of fast nuclear systems, taking into account present perspectives for deployment of advanced thermal reactors and future FNRs. Similarly, the evaluation of the impact of these technologies on the DGR designs, taking into account updated nuclear policies of EU Member States, technology deployment and different options for fast systems deployment, needs still to be completed.

Scenario studies, including industrial implementation aspects and, possibly, economic evaluations, should take into account the combination of various reactor types, including FNRs or ADS, in order to identify potential synergies, including with current LWR parks. Furthermore, these scenarios should allow indicators for decision making that include all aspects of the problem to be quantified: overall safety, consumption of natural resources, nuclear material inventories to be managed, environmental impact, costs, reactor fleet composition, time projection to reach equilibrium, industrial capacities required for fuel treatment and fabrication (including MA bearing fuels or targets), technical difficulties, secondary waste generation, occupational exposures, proliferation concerns, need for (cross border) transport, public acceptance, etc.

The feasibility of alternative fuel cycles should be investigated also for HTR fuel, which could become the mainstream technology for nuclear cogeneration. These alternative fuel cycles may comprise transuranic and minor actinide recycling, use of thorium fuel to stretch the resource base and to reduce minor actinide production, plutonium burning and deep-burn concepts as well as core/fuel management to optimise the cycle length between fuel reloads/shuffles.

Finally, it should be mentioned that currently there are no short or medium term industrial prospects in Europe for the deployment of the thorium cycle. However, thorium could become an attractive option for the long term due to its large European resource base and potential role in the nuclear waste minimisation.

An interesting strategy for the long term could be the combination of Molten Salt Reactors

(MSR) technologies, both with thermal and fast neutrons, with the thorium fuel cycle. MSR, by its intrinsic reutilisation of fuel in the reactor, could allow using all the thorium natural resources, in an efficient way, while simultaneously reducing by large factors the production of transuranic elements. R&D on MSR, and its utilisation within the Th cycles, is required to clarify the feasibility and potential benefits of MSR and of the Th fuel cycle.

The SNETP vision on R&D needs for both MSR[15] and the Thorium cycle[16] are unchanged from that documented earlier. These annexes describing those R&D needs are still valid and supported by the present SRIA. They show the significant long-term potentialities and the significant challenges to make industrial implementation of these systems and the associated R&D priorities.

15 - SRA Annex: Molten Salt Reactor Systems http://www.snetp.eu/www/snetp/images/stories/Docs-SRA2012/sra_annex-MSRS.pdf

16 - SRA Annex: Thorium cycles and Thorium as a nuclear fuel component http://www.snetp.eu/www/snetp/images/stories/Docs-SRA2009/sraannex3final.pdf

NUGENIA - nuclear fission technologies for Generation II and III nuclear plants

1. Introduction

NUGENIA ("NUclear GENeration II & III Association"), established on 14 November 2011, developed the roadmap which forms the basis for the Generation II/ III part of the SNETP SRIA. The main mission of NUGENIA is to be the integrated framework among industry, research and safety organisations for safe, reliable and competitive Generation II & III fission.

Figure 14: Golfech NPP over the Garonne river (Source: EDF)

SNETP mandates NUGENIA to act as the body in charge of coordinating at EU level the implementation of the R&D within this technical scope. Under this mandate, NUGENIA is in charge of the following activities:

- Defining detailed roadmaps and R&D priorities
- Facilitating the emergence of projects implementing R&D in the field of Generation II & III
- Identifying all relevant funding sources for Generation II & III R&D
- Generally promoting European nuclear Generation II & III collaborative R&D
- Facilitating cooperation with international counterparts on Generation II & III R&D

In order to gather and share the best available knowledge, skills, facilities and technologies, the work of NUGENIA is organised in six technical areas:

1. Plant safety and risk assessment
2. Severe accidents
3. Improved reactor operation
4. Integrity assessment of systems, structures and components
5. Fuel, waste management and dismantling
6. Innovative LWR design

plus two cross-cutting areas:

- Harmonisation
- In-service inspection and inspection qualification

The roadmap presented in the following subchapters is based on extracts from the detailed NUGENIA roadmap of the 6 technical areas above. Harmonisation topics are described in this introduction and the in-service inspection and qualification topics are addressed in technical area 3.

A transverse objective of NUGENIA is the harmonisation, which is aimed at reducing any substantial difference within a group of countries in design and fabrication of systems and components, in nuclear safety level and requirements, as well as safety assessment processes and practices. It involves the search for a long-term convergence towards the agreed WENRA objectives and the shared way to achieve them.

The objectives of the NUGENIA Harmonisation Cross-Cutting Area are therefore: supporting the deployment of nuclear energy within the European market setting-up the basis for an effective standardisation of reactor component assessments; improving the safety level of the nuclear installation through shared design approaches and licensing processes.

Three main fields of endeavour are likely to support R&D programmes:

- acquisition of data, through pre-normative research PNR for new design and operating conditions, but also for definition of operational limits, establishment of practices and safety criteria
- establishment of shared codes and standards
- search for strategies providing with smooth and efficient methods to enlarge progressively the field of consensus among stakeholders

In the design of new plant or systems, the adoption of advanced methodologies, combining defence-in-depth, risk-informed and safety margin approaches underlines the need for new research efforts.

Extended sharing of operating experience among stakeholders has to favour the adoption of common best practices

The objective for development of codes and standards - while keeping coherence with the already existing efforts - should be oriented towards:

- support to the development of EU standardisation and regulatory frameworks for the safe operation of the nuclear installations
- development of handbooks and codes of best practice relying upon European Standard Organisations

Extended sharing of operating experience among stakeholders has to favour the adoption of common best practices.

There process of harmonisation in the nuclear industry at European - and international - level is supported by industrial organisations, and the European regulatory authorities have defined safety objectives that are referred to in the EU Nuclear Safety Directive.

The harmonisation strategy is not aimed at issuing new safety directives, but at finding practical methodologies to reach the above mentioned safety objectives, for instance through appropriate safety criteria and practices. It should rely upon a systematic and continuous dialogue between the stakeholders.

NUGENIA gathers those stakeholders that adopt the regulations and rules for design, operation, safety-survey and safety-assessment. Thus it appears as a favoured place to create harmonisation through scientifically funded consensus. That way it can complement the activity on standardisation already going on in other organisations.

TECHNICAL AREA 1 - Safety and risk of NPPs

i - Scope and objectives

Technical Area 1 is devoted to evaluating the risk caused by the existing NPPs during their operation up to situations with core degradation, therefore developing and optimising the use of methodologies to evaluate their safety level. This implies improving the assessment of numerical simulation uncertainties and of safety margins.

Due to the general reduction of nuclear risk through improved system safety features, reduced parameter uncertainty, and better evaluation of the nature and intensity of aggressions and hazards, the causes of residual risks become comparatively more important.

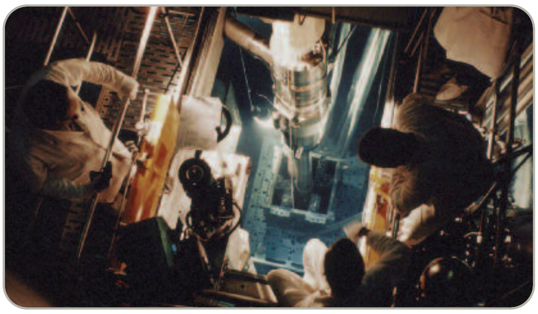

Figure 15: Phebus reactor pool and the experimental reactor core down 8 meters (Source: IRSN)

This residual risk is mainly originated from:

- the increasing heterogeneity among a plant fleet, due to the generic improvements and the integration of the R&D outcomes, which cannot be performed at the same time in all the plants of a fleet
- the non-compliance of the plants with the reference state, which is increasing with the life, due to the build-up of intervention and manipulation errors, the non-conformity of system and material assembling, the unavailability of spare parts

- the a priori assumptions in the modelling, such as symmetry and homogeneity, and the errors in the design data-set computation

Those random components of the risk can be neither easily investigated nor detected, so that it must be considered as a residual occurrence in the safety demonstration.

SNETP should play a driving role in supporting these approaches which can contribute significantly to improving the safety assessment.

In order to overcome the current limitations on accuracy, other safety evaluation methodologies such as those relying upon decision-making theory and/or economic models, but also risk-space based methodologies should also be investigated and evaluated.

In practice, the following sub-areas have been identified to progress upon:

- Contributing to further development of common understanding and usage of risk assessment techniques based on probabilistic safety assessment (PSA). In the field of input data, a special attention will be devoted to the evaluation of different kinds of dependency and human performance effects, and associated reliability data.
- Improving the deterministic assessment of plant transients with conservative assumptions and extended coverage of validation and extending consensus on methodologies for transients' evaluation.
- Improving methodologies to assess the impact of external loads and hazards on barriers and on structures, systems and components. Following Fukushima lessons, the effect of both single and multiple external events on safety function degradation need to be considered. Time (and feasibility) for recovery and the influence of non-safety systems on barrier strength are important to position.
- Among external events, a special focus will be put on the impact of electrical disturbances from the grid on plant safety systems and safety functions: the objective is to secure safety system performance.
- Integrating the deterministic and probabilistic safety assessments in order to better evaluate safety margins with best estimate methods, in particular determining the data, methods and knowledge needed to assess safety margins in components such as pipe/vessel and system (strength and weaknesses of NPPs).
- Designing of reactor safety systems to handle obsolete components and to up-grade them to handle new safety demands.

ii - State of the art

So far the activities have used inputs from international organisations, EC Framework Programme, and international initiatives; the main reference programmes in the safety evaluation are:

- for probabilistic methods the main programmes are OECD WG-risk, NPSAG, SAFIR, VGB-PSA working group, APSA-network, ASAMPSA-2, and MMOTION
- for deterministic assessment of plant transients: OECD WGAMA, OECD LOFC, OECD PKL, OECD ROSA, OECD SETH, NORTHNET, SAFIR, VGB, NURESIM, NURISP, NURENEXT, BEMUSE, EUROSAFE, AER, OECD ISPs…
- for impact of external loads and hazards on the Safety functions: NORTHNET, SAFIR, OECD PRISME and OECD PRISME2, NOG, VGB, ASAMPSA2-E, NPSAG
- for impact on the safety functions of external electrical disturbances: OECD DIDELSYS, BWR-club, NPSAG, IAEA (NS-G-1.8, D-NG-T-3.8)
- for advanced safety assessment methodologies: OECD WGAMA (PROSIR), NORTHNET, SAFIR, EU-RTD-programme (NURBIM, NURISP), OECD BEMUSE, OECD SM2A, OECD UAM and OECD PREMIUM

iii - Challenges

1.1 Challenges in the field of PSA methods

1.1.1 – Quantitative aspects of PSA

For quantitative aspects, the following gaps should be covered: development of methodologies to quantify initiating event frequencies for low probability events, including external events and Common Cause Failures (CCF) events, combination of events (including internal and external events), methods used for establishing component failure rates with focus on components with low failure rates and also failure rates due to specific loads (such as loads due to fire, severe accident conditions…), data and methods used to assess CCF inside specific system and trans- systems interactions at component level and at subcomponent level, development of methods for assessing human reliability and establishing a database with reference cases for supporting common risk assessments of human performance, methods to handle time dependent assessments in PSA, methods and data for quantifying the effects of aging on PSA-outputs, methods and data for assessing failure frequencies of digital components, methodology and data for performing fire PSA.

For the PSA dedicated to source term issues (level 2 PSA) recommendations on the best strategies to couple level 1 and level 2 PSA should be done, and methodologies to assess shut-down state or external events should be developed.

1.1.2. General aspects of PSA results

The R&D topics providing estimation of overall risk of plant operation are:

- considering several plants simultaneously affected notably in case of external events
- benchmarking of existing PSA-studies to support comparability of PSA studies and to support use of safety goals in plant management
- developing guidance to use safety goals in reactor safety assessments
- assessing of risk related to spent fuel pool
- modelling techniques for functional dependencies in electrical and safety instrumented systems
- administration of PSA models and related documents, including review, considering recurring updates and use of special models for separate analysis in support of various safety related assessments due to changes to the licensing basis
- establishing methodologies for level 3-PSA (up to health effects) including integration with level 2-PSA

1.1.3. Best practice for PSA application

Even if quite advanced PSA tools are well developed and extensively used, it requires also applying them correctly and in the most useful manner in specific Risk Informed Decision Management (RIDM) applications. If good methodology for development of "basic" PSA is necessary the need of a large practice of well-developed and harmonised RIDM methodology is even more necessary.

1-2 Deterministic assessment of plant transients

In the assessment of plant transients, new challenges are arising but recurrent studies still need large efforts to be performed with effective results. Indeed, plant behaviour is the result of complex multidimensional physical phenomena; moreover, phenomena are tightly coupled:

1.2.1. Improved thermal hydraulics evaluation for the existing plants

To fulfil these goals, the main challenges in thermal-hydraulics are:

- a better understanding and modelling of the multi-dimensional phenomena, in particular in vessels and pools
- a better understanding and modelling of the multiphase (steam/water, non-condensable gases…) phenomena
- the interaction with neutronic (in particular in reactivity transients), mechanic (fluid-structure interaction in steam generator for example), and thermo-mechanic
- the experimental validation
- the uncertainties evaluation

Some specific topics are given hereafter:

- stratification in pools and vessels, in particular for BWR- condensation pools when steam flow is low
- mixing in pools and vessels, in particular at low flow rates, including vessel pressurised thermal shock and boron mixing in reactor vessels
- better predicting of the margins of instability in BWR-cores, in particular coupling 3D thermal-hydraulics with neutronic codes
- 3D flows in the reactor pressure vessel. (BWR/PWR/VVER)
- assessing effects of non-condensable gases in pipes for scenarios with gas intrusion

1.2.2. design and evaluation of passive safety systems

The major challenge to a generalised adoption of passive systems for safety purposes is the achievement of a convenient and exhaustive full scale demonstration of their reliability in transient conditions. Specific R&D is to be devoted to provide evidence of the system reliability despite the approximations and assumptions in the validation experiments and to clear the way to extrapolation.

1.2.3. Coupled multi-physics codes,

Couplings such as neutronics and thermal-hydraulics have to be developed, for example for re-criticality scenarios in which several control rods partially scram (BWR).

1.2.4. Containment behaviour

Particular phenomena need to be better modelled:

- non condensable gas flows in the containment with and without spraying
- heat transfer in the gas phase of the containment including the interaction with walls and pipes
- leak rates through containment up to containment break
- clogging phenomena in strainers and fuel and associated assessment methodology
- fire and gas explosion simulation methods and applications to reactor safety

1.2.5. Fluid structure interactions

Special models need progresses:

- turbulent flows and its effects on component aging
- fluid-structure interaction in steam-generator
- water hammer assessment
- coupling between CFD and system codes
- heat transfer along piping and vessel walls during turbulent flows

1.2.6. Fire risk

- comprehensive characterisation of different fire loads
- fire suppression models and suppression technologies
- methods and criteria to assess malfunction of electrical equipments considering combined effects of soot and thermal stress

1-3 Impact of external loads and hazards on the safety functions

The external events have to be characterised by loads and frequency as well as by the risk for co-incident occurrences and the effects on non-safety systems on the safety system. The potential for successful preventive and mitigating human actions has to be considered. Methods for frequency/magnitude assessment for events with short and long return periods (from 100 to more than 1000 years) need to be further developed in view of the major uncertainties involved. Estimates of the effects of climate change also indicate substantial impacts on the frequency and magnitude of certain natural external events even in the near future, which needs to be considered in the analysis of external events.

Methods and methodologies to identify single and multiple external events are also necessary to assess effects on a multi-unit plant, as well as how the effects of external events on non-safety systems could affect safety systems.

1-4 Impact of external electrical disturbances on the safety functions

Among the external hazards, particular attention has to be paid to grid disturbance effects on the plant through the internal electrical buses and other electrical components important to safety. It includes assessment of the effects originating either from lightning or from motor magnetic fields on modern electronics and/or digital equipment which are far more sensitive to magnetic fields than components used in the past. Other equipments like plant electrical and I&C equipment, or diesel generators for emergency power may be also affected. The design of plant control and protection systems has to be based on an increased understanding of these effects and the sources of these effects have to be investigated.

1-5 Advanced safety assessment methodologies

The general challenge in this sub-area is to increase knowledge about the existing safety margins of a plant. Several methodologies can be further developed in order to improve the accuracy of evaluations:

- Risk informed methodologies, usually developed by operators, based on decision-making theory and/or economic models can still be improved in order to optimise the risk accuracy evaluation,
- Understanding the safety margins and best estimate methods, i. e. integrating the deterministic and probabilistic safety assessments, is another possible route to improving accuracy. This includes development of methodologies (such as Dynamic PSA and Monte-Carlo assessments or a combination) are used in parallel with existing methods for probabilistic and deterministic assessments.

A particular challenge for these methods is to better model/predict dynamic behaviour and also time dependent scenarios while not affecting the assessment of uncertainties.

Application of these developments should address better evaluation of safety margins for RPVs, main containment, passive system and pipes in case of LOCA or PTS events, but also for beyond design situations or natural circulation conditions.

1-6 Design of reactor safety systems

Among the challenges for the design of safety systems the most important are:

- design of digital system with integration into existing plants
- increased diversification and robustness of safety systems
- use of passive system for safety function
- methods for reactivity measurements under accident conditions
- design of level measuring systems to withstand high temperatures

TECHNICAL AREA 2 - Severe accidents

i - Scope

The main public safety goal for nuclear power is to prevent a societal calamity and huge economic loss. With appropriate site risk evaluations, plant designs and management, current Generation II and future Generation III nuclear power plants (NPP) show high levels of robustness and low probabilities for severe accidents (SA). But, despite the highly efficient accident prevention measures adopted for the current Generation II and the still more demanding ones for the Generation III plants, some accident scenarios may, with a low probability, result in SA, as recently emphasised with the Fukushima Daiichi accidents in Japan. This SA can result in core melting, plant damage and dispersal of radioactive materials outside of the plant containment, thus threatening public health and the environment.

This risk can be substantially decreased when state-of-the-art devices currently available for prevention and mitigation of severe accidents are installed. Lessons from the Fukushima accident and consequences related to Accident Management provisions from the recently completed ENSREG stress tests and other national activities will lead to further enhancement of the safety of NPPs.

Within this technical area, general objectives are defined and followed by specification of research and innovation challenges to further reinforce NPP safety provisions.

ii - Objectives

Some predominant phenomena require a better understanding in particular to improve Severe Accident Management Guidelines (SAMGs) and to design new prevention devices or systems for mitigation of SA consequences.

Seven technical sub-areas address the 6 objectives listed below, the three first ones being directly linked to mitigation processes:

- in vessel corium/debris coolability
- ex-vessel corium interactions and coolability
- reducing source term
- impact of severe accidents on the environment
- severe accident scenarios
- improving the emergency preparedness

New experimental efforts will be needed in most sub-areas accompanied by modelling development and validation.

The knowledge gained and the modelling improvements will allow the optimisation of SAMGs and the improvement of prevention and mitigation systems such as core reflooding systems, filtering systems, venting systems or recombiners in the containment.

iii - State of the art

Considerable knowledge has been gained about SA phenomenology through studies carried out during the last 30 years. It is based on experimentation, mostly out-of-pile, with a few in-pile programmes like Phébus FP, and theoretical simulations, as the accidents at TMI2 in 1979 and at Chernobyl in 1986 were the only major NPP reference cases until the Fukushima Daiichi accident.

Since 2004, the state of the art is periodically updated in the frame of the SARNET network (Severe Accident Research NETwork of Excellence), coordinated by IRSN in the 6th and 7th Research Framework Programmes of the European Commission (see www.sar-net.eu). In particular, the ranking of R&D priorities has been recently reviewed to take into account early feedback from the Fukushima Daiichi accident (and such update will continue in the next years according to the progress of the understanding of this accident). The identified challenges account also for the results of all past and ongoing international programmes (Euratom-FP7, OECD/NEA, ISTP…).

iv - Challenges

The highest priority safety challenges are described in the following sub-areas: In-vessel corium/debris coolability, Ex-vessel corium interactions and coolability, Containment behaviour including hydrogen explosion risk, Source term, SA impact on the environment, and emergency and preparedness management. One transversal sub-area concerns the SA scenarios.

2.1 In-vessel corium/debris coolability

Substantial knowledge exists concerning cooling of intact rod-like core geometry. The main challenge for long term R&D will be to address the remaining uncertainties concerning the efficiency of cooling a degraded reactor core, with presence of corium and/or solid debris, by water addition to limit or terminate the SA in-vessel progression.

The impact of the further analysis of the Fukushima accident will be taken into account, and, conversely, R&D will be important for the plants decommissioning.

R&D topics

- **The highest priority R&D topics for this challenge are: debris bed formation and cooling; for in-vessel melt retention, corium pool coolability in the reactor pressure vessel (RPV) lower head, especially for BWRs with presence of control rod and instrumentation guide tubes; critical heat flux and RPV external cooling conditions.**

2.2 Ex-vessel corium interactions and coolability

For ex-vessel corium situations, the major safety challenge is to preserve containment integrity against rapid failure (steam explosions, Direct Containment Heating or DCH) or slower basemat melt-through (by Molten-Core-Concrete-Interaction or MCCI) and/or containment over-pressurisation.

R&D topics:

- **The highest priority R&D topics for this challenge are: fuel-water premixing and debris formation and coolability; complementary MCCI research to cover oxide-metal layer interaction and all reactor concrete compositions; assessment of MCCI top flooding; and finally analytical work to transpose MCCI experiments to reactor scale.**

2.3 Containment behaviour, including hydrogen explosion risk

The containment represents the ultimate barrier to prevent or limit the release of fission products (FP) to the environment during a SA. If local concentrations of combustible gases (hydrogen and carbon monoxide) are present, gas combustion might occur and cause a pressure increase that could eventually lead to containment failure.

Efforts in the short and mid-term should focus on extensive simulations of gas distribution in the presence of mitigation systems using both Lumped-Parameter and CFD (Computational

Fluid Dynamics) codes in order to interpret a whole set of different experiments with consistent models. Reliable models of deflagration and deflagration-to-detonation transition should be developed in order to improve the present modelling mainly based on empirical correlations.

R&D topics

- The highest priority R&D topics for this challenge are containment atmosphere mixing (including BWR containments with nitrogen atmosphere) and gas combustion, which imply the following phenomena: gas distribution in the containment with the influence of mitigation systems, pressure increase during hydrogen combustion, and deflagration to detonation transition. Scaling (qualitative and quantitative) of phenomena from experimental facilities to actual containments should also be addressed with priority.

2.4 Reducing source term

The source term to the environment refers to the amount, chemical speciation and isotopic speciation of all radio-elements that can be released to the environment. At present, the increased safety requirements in both existing plants and new designs aim at reducing the source term by proper measures for limitation of uncontrolled leaks from the containment and for improvement of the filtering efficiency of containment venting systems.

The Fukushima accident underlined the need for studying the impact on the source term of the filtered containment venting systems which are important radionuclide-removal processes.

R&D topics

- The highest priority R&D topics for this challenge are: impact of filtered containment venting systems on source term and development of improved devices; oxidising environment impact on FP release from fuel, in particular for ruthenium, i.e. under oxidation conditions or air ingress for high burn-up and MOX fuels; high temperature chemistry impact on FP behaviour in the Reactor Cooling System (RCS), i.e. improving predictability of iodine species exiting RCS towards the containment; containment chemistry impact on source term, i.e. improving the predictability of iodine chemistry in the containment.

2.5 Impact of severe accidents on the environment

The SA impact on the environment in the near-field around the NPP must be assessed as part of the Environmental Impact Assessment (EIA) of an NPP in accordance with European and national legislation. Here only the atmospheric dispersion of radio-nuclides is addressed. This will allow an interface between PSA Level 2 and assessment of radiological consequences and it will improve the emergency planning and zoning and the post-accidental situation management.

Research efforts on atmospheric dispersion modelling over recent decades have produced several models; some of them are already used in the preoperational or operational framework in case of radiological emergency. Despite this effort, some issues still need to be tackled in the preparedness phase, in the response phase (to anticipate properly the situation in order to protect the workers and the surrounding population in time), and in the post-accident phase.

Strong links should be established in the future with the NERIS platform (European Platform on Preparedness for Nuclear and Radiological Emergency Response and Recovery: see www.eu-neris.net).

R&D topics

- The highest priority R&D topics for this challenge are: accurate atmospheric dispersion models, in particular accounting for chemistry of radio-elements (integrating on-site experiments and use and development of specific CFD models); accurate evaluation of source term to the atmosphere (by using inverse atmospheric modelling and comparisons with the environmental measurements); adaptation and development of "ensemble computations" (i.e. variations of calculations for different initial weather data) for atmospheric dispersion of radio-elements from NPP.

2.6 Severe accident scenarios

Integral codes (or system codes) are essential for simulating all SA scenarios including the evaluation of the source term into the environment, as well as the evaluation of SAM measures and the efficiency of mitigation systems. The highest priority is to continue to capitalise on knowledge gained from using these codes, particularly the ASTEC code (IRSN-GRS), and to feedback the interpretation of the Fukushima accident in the coming years. Attention should be paid in particular to models of BWR core degradation and to their validation.

In addition, the Fukushima accident has underlined the importance of the behaviour of spent fuel pools in case of loss of cooling and the need of new SA instrumentation for SA diagnosis and management, as well as for early source term predictions and emergency preparedness outside the NPP site.

Another essential issue is the need to store in reliable and durable databanks the results of the huge amount of SA experiments that were performed over more than 30 years. They should remain available for any further analysis of SA phenomena and for validation of simulation codes.

R&D topics

- The highest priority R&D topics for this challenge concern the continuous capitalisation of knowledge in the integral codes, particularly ASTEC, and the improvement of their applicability to spent fuel pools. The latter will need further R&D on the following phenomena: large-scale flow convection, impact of partial dewatering of fuel assemblies on thermal runaway and fuel degradation, clad and fuel mechanical behaviour in an air-steam atmosphere. Other challenges will be to investigate on one hand new SA instrumentation and on the other hand critically risks in case of spent fuel pool dry-out or of damage NPP core.

2.7 Improving the emergency preparedness and response

In the emergency preparedness and response area, an accident can be roughly broken down into two phases: the emergency phase (in red on the diagram) and the post-accidental phase (in green). The post-accidental phase is the focus of another project integrated in the NERIS SRA, and the scope dealt with in this paragraph is the emergency phase, itself composed of a threat phase and a release phase.

Between the onsite emergency declaration (reactor out of normal operation) and the release of a radioactive plume, the main responsibility lies with the operator who has to take actions which may stabilise the plant and/or mitigate the consequences of the accident. The longer this phase, the more time there is to implement recovery action and the better are measures that can be taken to protect the population around the plant.

Capabilities to overcome the accident require the availability of reliable information, the capacity to operate correctly under high stress and harmful conditions, and to dispose of efficient and reliable tools.

Reliable information comes first from the plant, in which the main systems for safety have to be backed up, including with externally diversified sources, but reliable information needs also reliable instruments and good interpretation in order to discard spurious signals. In that prospect, human and technical redundancy is advisable.

Operating correctly in high stress conditions needs first to know how to behave as a function of the plant and how to recover from specific situations. Indeed, the correct operation needs a fully developed planning phase (see sketch above) with severe accident management strategies, plans and procedures to cope with short term reflex actions and accurate diagnosis capabilities to prepare prognoses corresponding to different worsening hypotheses on future evolution. This asks for preparedness in terms of scenarios, information, fast and reliable computation, adaptation capacities, and technical systems not destroyed by the ongoing accident.

For meeting that objective, redundancies are also useful in terms of technical means and evaluation teams in order to secure the diagnosis and prognosis.

Last but not least, reliable tools encompass, well exercised information systems and emergency organisations, instrumentation, fast computer systems to study multiple future scenarios, highly trained and complementary teams with experts in informatics, reactor physics, reactor operation, meteorological data, and integration of all the different aspects.

R&D topics

- In the domain of research several fast running computer systems are already available, but several issues still remain open, mainly in the field of human behaviour and organisation. Technical means and trained personal are systematically developed by the operators with support from the local organisation for civil security. The STATIC programme studies a part of the human and organisational factors during crisis, but organisation-dealing aspects still need to be improved: it appears useful to improve knowledge on human behaviour under stress conditions, to extend redundancy among teams without slowing down the engineering capacity; international cooperation, beyond existing reporting systems, it may be useful to share online data and support expertise; this requires evolution of the existing international conventions and formatting information to common standards.

Moreover, it should be worth:

- Providing the national Safety Authorities with technical support furnished with fully online updated data, developed technical means, and formatted exchanges with the operators, in order to create redundant expertise contributing to release threat resolution through advice to the safety authority,
- Extending general knowledge of SAM and capitalising it within fast running computing tools; benchmarks could be organised in that field, with respect to the possible source term evaluation, including the capacity to use environmental data to fit it to reality,
- Dispatching emergency mobile means for monitoring the releases,
- Integrating for each site the multiscale dispersion calculations with multiscale meteorological integration,
- Developing suitable monitoring in-situ and mobile devices,
- Enhancing the awareness of the socio-economical context (public acceptance, media coverage…),
- Capitalising - in guidelines and suitable computation tools - the outcome from the operating experience feedback, the periodical crisis exercises, the studies and R&D programmes carried out within national and international frameworks.

TECHNICAL AREA 3 - Improved reactor operation

i - Objectives and motivation

Safe and efficient operation of the plants is the result of a blend of human, organisational and technological aspects, and R&D in all these fields can play an important role in continuously improving operational practices. Important issues related to operation are also discussed in other NUGENIA areas; here the following aspects are discussed:

> Safe and efficient operation of the plants is the result of a blend of human, organisational and technological aspects

- Human and organisational factors
- Integration of digital technologies
- Core management
- Water chemistry and LLW management
- Radiation protection
- In service inspection and inspection qualification

The rationale for collecting these issues together is to try to integrate them into a common advanced vision of the operation, supported by the implementation of modern digital technologies.

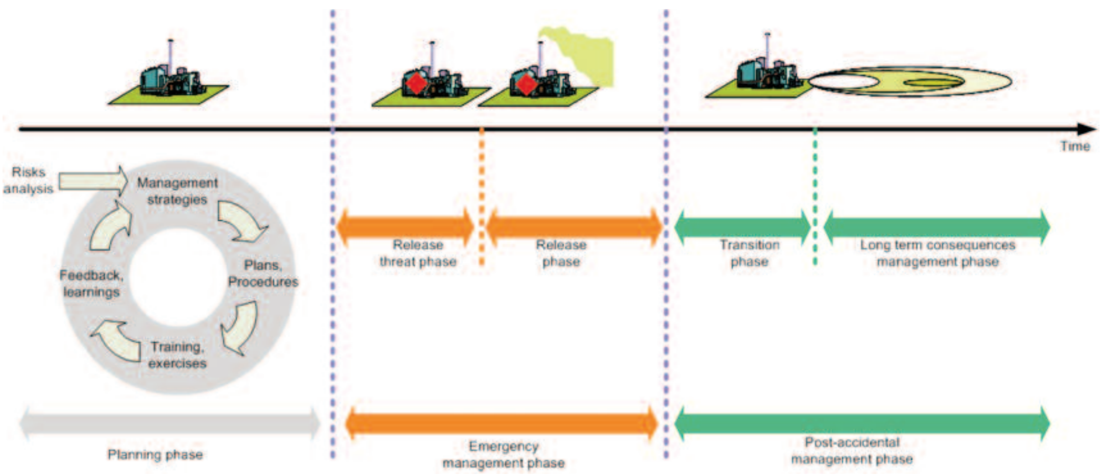

Figure 16: Emergency management phases (Source: IRSN)

ii - State of the art and challenges

3.1 Human and organisational factor

The Human and Organisational Factors (HOF) community is promoting R&D in various international connections e.g. in the OECD/NEA working Group of Organisational and Human Factors (WGHOF).

A road map in this field was recently developed by the partners of the MMOTION project. They proposed four research programmes:

- "Risk-informed decision-making in design and operation"
- "Culture and practices for safety"
- "Integrated design approaches"
- "Technological requirements in nuclear and other high risk industry"

The aspects which are considered as utmost priority in this field are evaluation of human reliability analysis, operational culture and work practices. Important challenges are to strengthen the objectivity of safety judgments by using methods of risk-oriented decision making in the human reliability area, to improve the effectiveness of safety provisions, to harmonise operational principles across Europe and to minimise the negative impacts of complexity on operation and safety.

R&D topics

- To develop advanced and harmonised methods, solutions and tools for evaluation of operating experience about human and organisational factors.
- To develop tools for risk-informed decision making support and manage socio-technical systems complexity at the design level.

Organisational safety culture and operating practices influence the safety level. Research should define the conditions required for ensuring the robustness of the organisations in charge of operating NPPs, based on a deep understanding of practices and culture and of change impact on the socio-technical system.

It would be particularly important to consider how individuals, teams and organisations function and interact within the plant within a specific safety culture, and how they are supported by tools, artefacts, procedures, rules.

R&D topics

- To develop theoretical models of the socio-technical system as well as effective methods and tools to benchmark operational practices and to assess human performance.

The Fukushima accident revealed significant weaknesses in the on-site and off-site response to extreme, unforeseen events, in particular when the functioning of the emergency organisation itself is weakened by the event.

R&D topics

- To understand human and organisational behaviour under high stress and harmful conditions.
- To develop reliable solutions for inter-organisational collaboration (including plant, utility, emergency services, regulator and government agencies), and for taking time-critical decisions based on dynamic and partially unreliable information.

3.2 Integration of digital technologies

Digital technologies are nowadays deployed in all modern power generation plants and also in large industrial. The situation in the nuclear power sector differs from other sectors in the following key aspects:

- the use of analogue systems is being extended beyond their initially expected service lifetime
- regulatory uncertainty and associated financial risk concerns are delaying the deployment of already available new technologies

Implementation of digital technologies is a key issue for the life extension of Generation II reactors, as well as for the deployment of Generation III, because it offers an unique opportunity for improving operational performance in respect of safety margins. Several European plant operators have gone through relevant modernisation programmes, often performed under the constraint of both minimising the impact on the traditional plant operating, managing and maintaining modes and practices, and searching of solutions to immediate needs. Due to the fast evolution of these technologies, this approach is not sustainable in the long term.

In USA a huge research programme is being set up, intending to develop standards and guidelines to facilitate the transition to digital technology and its deployment across the USA nuclear fleet, potentially influencing the R&D activity also in Europe.

R&D topics

- To develop a new advanced digital information and control architecture to integrate all plant applications, respecting safety constraints, and to collect and organise data from all types of sources of condition-monitoring data.
- To provide technical studies and guidelines for a wide implementation of digital technologies.

In parallel to the use of programmable digital electronic systems in nuclear safety applications, to overcome the difficulty of maintaining analogue electronic assemblies and to take advantage of functions enabled by digital logic, attention is needed regarding the increase of the potential for component-level faults due to engineering mistakes.

R&D topics

- To develop methods for analysis of the safety software to ensure the reliability of the software for regulatory assurance and the safety of instrumentation and control systems.

To achieve better integration of technical, human and organisational factors, more focus is needed on the efficient integration of HOF in the design of control rooms and on assessing the future working conditions through the active participation of HOF specialists and end users.

R&D topics

- To develop engineering procedures for integration of human and organisational factors in to the overall design process.
- To develop advanced alarm systems capable of improving operator performance, permitting better recognition and comprehension of alarms.

Several kinds of mobile communication/computing devices may be used to provide continual plant status and control capability anywhere in the plant, improving the performance of field workers. Tools using virtual reality models and technologies could be used to develop computer aided maintenance procedures and to train maintenance personnel.

R&D topics

- To develop suitable procedures and advanced tools to support field workers and maintenance operators.

3.3 Core management

Core optimisation, based on increased fuel utilisation and on a more accurate evaluation of the safety core characteristics, is achievable through the continuous improvement of the design and analysis tools, as well as through the improvement of the monitoring instrumentation.

This task can be translated into large challenges in basic nuclear data, neutronics, material science, thermo hydraulics, fuel fabrication, reprocessing and partitioning. Coupling all these aspects (multiphysics) and assuring modern quality software are the drivers to replace the current suites of simulation codes. Better accuracy has to be justified either against experimental data or against benchmark calculations.

R&D topics

- To enhance core modelling capability using the modern methods of calculation of power distributions and reactivity.
- To collate operational data from the NPPs to validate modern method of the calculation.
- To develop method for evaluation of uncertainties for core state in abnormal conditions.

Advanced instrumentation and measurement methods, and efficient signal analysis, can increase reliability, performance and competitiveness

Nuclear instrumentation is still mainly based on safe but conservative technologies. Present and future competitiveness depends also on the accurate and predictive knowledge of core behaviour. Advanced instrumentation and measurement methods, and efficient signal analysis, can increase reliability, performance and competitiveness.

R&D topics

- To explore, develop and define new strategies and approaches in core monitoring.
- To estimate and reduce measurement uncertainties.

3.4 Water chemistry and LLW management

Water chemistry management has the main scope to optimise the primary, secondary and auxiliary cooling systems chemical parameters.

Water chemistry has been the subject of large basic and applied research programmes, in particular in USA under the EPRI coordination. EPRI continues nowadays to update water chemistry guidelines, as well as to update optimisation tools to mitigate corrosion, to achieve and maintain fuel performance standards and to minimise radiation fields in the plant.

Water chemistry strongly influences the operational safety of reactors by affecting formation of deposits, which may cause heat transfer degradation and enhance localised corrosion. A good control of water chemistry can significantly reduce various operational problems, including corrosion, erosion, deposition of corrosion products, hence increasing the life of systems and components. This mainly consists of two essential parts:

- chemicals added to adjust the pH to minimise the corrosion rate of structural materials
- mitigation of the concentration of chemical impurities, which catalyse the degradation of materials, coolant and protective oxide coatings

R&D topics

- Development of cost-effective chemistry optimisation tools and techniques to improve plant availability and safety.
- Development of dedicated software tools to improve chemistry control and diagnostic capabilities.

The pressure to reduce the radiation exposure of the workers as well as the radioactive release into the environment requires constant improvement of processes and technologies for LLW treatment and for conditioning of liquid waste, potentially able also to reduce the costs.

Priorities in the treatment of liquid radioactive waste are to obtain higher decontamination and volume reduction factors, lower on-site processing costs, reducing solid radioactive waste generation rates and minimise corrosion.

R&D topics

- To develop advanced filtration and innovative removal processes to monitor, control and limit activated corrosion product transport.
- To look for materials with lower content of cobalt, and with lower cationic release rates under LWR water chemistry conditions.
- To study updated procedures for the chemical decontamination of surfaces.
- To develop updated systems for tritium, carbon ^{14}C and boron management.

3.5 Radiation protection

The strong attention for radiation protection has recently led to the establishment of the Multidisciplinary European Low Dose Initiative (MELODI) platform, with the aim of better understanding the health effects of exposure to low dose ionising radiation. In 2011 this platform issued a Strategic Research Agenda, which defines a series of topics suitable to be considered in a long-term perspective.

There are however several topics in the radiation protection area which can be appropriately considered in the framework of NUGENIA, since they are very tightly related to the everyday plant management activities.

R&D topics

- To develop new technologies and tools for managing information related to the radiological dose to plant workers.
- To develop improved tools for radionuclide release minimisation and for the analysis of the consequences on the environment.

3.6 In service inspection and inspection qualification

In-service inspection (ISI) of nuclear power plants (NPP) by non-destructive testing (NDT) is a very important part of the NPP ageing management or predictive maintenance. R&D topics in this field are mainly focused on qualification of NDT systems and risk-informed in-service inspection (RI-ISI) methodologies, with the objective increase their effectiveness and efficiency and to respond the new challenges resulting from NPP's long term operation and new build.

The European Network for Inspection and Qualification (ENIQ) is recognised as one of the main contributors to today's global qualification situation along with other initiatives like the PDI on the United States of America. The inspection qualification methodology developed by ENIQ is also accepted by the IAEA as recommended practice to be followed for nuclear inspection qualification all over the world.

The main challenges for NDT qualification are on the new procedures based on phased array ultrasonic testing, time of flight diffraction ultrasonic testing, computed radiography, as well as the extension of applicability to other methods.

Qualification of NDT systems should enable efficient structures and components condition monitoring by NDT in order to improve maintenance of plants.

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R&D topics

- Support extended use of computer modelling and simulations.
- Develop a recommended practice for computed radiography qualification and Phased Array qualification under ENIQ type Methodologies.

The present challenges on the RI-ISI field are on pre-service inspection (PSI) for new build, risk reduction quantification and optimising ISI frequency.

R&D topics

- Analysis of the role of RI-ISI in defence-in depth and development of procedures to determine the achievable level of risk reduction.
- Development of RI-ISI and qualification guidance for a new build, for non-pressure boundary items and for the use of expert judgement.
- Development of alternative methods and guidance (to ISI) for managing risk.
- Produce specifications for quantification of Probability of Detection (PoD), including Monte-Carlo approach, and define the role of quantified PoD in risk reduction.

■ TECHNICAL AREA 4 - Systems, Structures and Components

i - Scope + Objectives

Structural assessments are an important part of NPP management programmes (e.g. ageing management, maintenance and design changes). These assessments are required for the effectiveness of periodic safety reviews. Aspects that need to be considered include definition of integrity assessment over the whole life cycle, the various degradation mechanisms, ageing issues, safety margins and harmonisation issues.

The Structures, Systems and Components (SSC) that need to be considered are those important for safety and availability, those that require high costs to replace and those that cannot be replaced without a significant long term refurbishment programme. Components like Instrumentation and Control (I&C) which are of high safety significance also need to be addressed.

While the assessment principles relating to SSCs are generally comparable in Europe, the actual methodologies and codes are different in the various European countries. With the longer term objective of European harmonisation in mind, it is necessary for the differences to be fully understood and for the lessons learned from Generation II nuclear power plants (NPPs) to be taken into account when developing and/or revising best practice guidance for the safe operation of SSCs with satisfactory, but not over-conservative safety margins. This is required in order to ensure high integrity and high performance in the case of internal and external loads. Post-Fukushima lessons imply that investigations of beyond design loads are also required to be considered.

This technical area roadmap comprises integrity assessment, description of loads, materials performance and ageing, ageing monitoring, prevention and mitigation, functionality and qualification.

ii - State of the art

Integrity Assessment of SSCs takes into account material properties, component geometry, loading and degradation mechanisms and effects, using essentially fracture mechanics methods. The assessment of for example RPV integrity assessment under Pressurised Thermal Shock (PTS) loading and in leak before break (LBB) analysis in piping is of critical importance to the safe operation of a nuclear power plant. Codes, standards and procedures are commonly used for integrity assessment. These are generally well founded and validated but in many cases they can have inherent levels of conservatism, particularly when considering plant life extension.

iii - Challenges

4.1 Integrity Assessment

There is a need to properly understand the levels of conservatism in the current integrity assessment methods with a view to revising the guidance and procedures. Such aspects as effects of load history, crack arrest, treatment of thermal and weld residual stresses and warm pre-stressing effects need to be considered with regards to this aim. Lessons learnt from Generation II NPPs in terms of integrity assessment validation should be considered and implemented.

> There is a need to properly understand the levels of conservatism in the current integrity assessment methods with a view to revising the guidance and procedures

The modelling of integrity assessment is important in order to be able to translate the mechanistic understanding to simulation tools and assessment procedures to predict theoretical margins for the safe operation of NPPs, taking into account structural features, real or postulated flaws, loading conditions and relevant material characteristics including ageing effects.

R&D topics

- Development of best practice procedures for assessing structural performance of multi-metal components, including cladded components.
- Development of a probabilistic approach on safety critical systems integrity assessment for long term operation and harmonisation of probabilistic safety assessment (in close link with Area 1).
- Treatment of non-crack like defects (Corrosion, thinning, pitting, erosion, flow induced corrosion, crevices).
- Integrity of RPV internals for long term operation considering also load effects like the dynamic response of reactor internals to Loss of Coolant Accidents (LOCAs); develop validated models for the assessment of structural integrity of in-vessel components under high doses of irradiation.
- Gain understanding of modelling approaches adopted in different European countries on life time evaluation of civil structures including creep behaviour and the influence of cracks.
- Develop probabilistic and deterministic methodologies to evaluate the impact of internal events (hydrogen explosion, pipe whip impact) and external hazards (seismic event, aircraft impact, explosion) on civil structures.

4.2 Description of Loads

Accurate knowledge of applied loads and resulting stresses (and strains) is needed for reliable SCCs. Increased computing power over recent years, coupled with advanced modelling capabilities, has resulted in enabling the accuracy to be evaluated in greater detail. Examples of this include piping system loads and stresses, stresses resulting from pressurised thermal shock loading and residual stresses resulting from fabrication processes such as welding.

R&D topics

- Development of guidance in order to more accurately predict fluid to component wall heat transfer (CFD, Computational Fluid Dynamics) for thermal fatigue analysis and fluid structure interaction under turbulent flow conditions.
- Establishment of the methodologies to rank external loads for deterministic and probabilistic assessments.
- Treatment of secondary and residual stresses.
- Investigation of combined fatigue and tearing fracture resistance under high asymmetry cycles and random high cyclic loads.
- Containment phenomena in dry-well like stratification and heat transfer into the condensation pool and load assessment from source to loads on walls under different voiding conditions in the condensation pool with the aim of reducing uncertainties.
- Increase the understanding of hydrogen production and hydrogen venting during service accident conditions and of hydrogen from zirconium oxidation and from radiolysis in water and hydrogen transport and absorption in water, gas phase and on civil structure walls.

4.3 Materials Performance and Ageing

The major challenge is to justify properly that all components affected by an ageing mechanism remain within the design and safety criteria. Indeed, properly understanding the performance of materials relevant to structural components and the effect of ageing mechanisms on their performance are key issues from the start to the end of life of each NPP.

Basically, ageing management should follow a well-defined procedure that consists of i) identification of the SSCs that are subject to ageing, ii) analysis, understanding and modelling of the main relevant ageing mechanisms concerning each SSC (potential or encountered) and finally iii) setting up of measures to justify the integrity of each SSC based on codes & standards, regulations, specifications & guidelines and scientific knowledge of the ageing mechanisms.

Figure 17: Reactor in Cadarache in French Provence (Source: EDF)

R&D topics

- An in-depth understanding of ageing mechanisms, determination of ageing.
- Close definition of damage rate and fitness analysis.
- Implementation of monitoring, surveillance and in-service inspection.
- Mitigation and repair measures taking into account the industrial capacity and obsolescence.

4.3.1. Ageing and Degradation Mechanisms

Ageing management addresses physical ageing that could result in degradation of systems, structures and components such that safety functionality could be impaired. Thus, degradation modes, including fatigue, irradiation embrittlement, stress corrosion cracking, irradiation assisted stress corrosion cracking, thermal ageing, general corrosion, erosion-corrosion, strain ageing, environmental fatigue, creep, creep-fatigue and thermal fatigue, need to be fully understood to ensure a good SSC status. It is therefore very important to be able to properly evaluate their positive or negative effect on "start of life" properties as they may become limiting factors for the safe and reliable operation of NPPs. Important as well is that the effects of the ageing and degradation mechanisms should be considered for the specific type of SSC material being assessed. Guidance and/or data are generally available for the projected lifetime but very scarce for allowing beyond design long term operation. Moreover, ageing management for long term operation has to take into account not only the physical ageing of materials but also the technological aspects (obsolescence). There is therefore clearly a need for further R&D to be undertaken in several of the areas referred to in order for a better fundamental understanding of the ageing and degradation mechanism to be realised and to lead to realistic assessment guidance.

R&D topics

- Investigation of microstructural and mechanical effects in RPV steels caused by long term irradiation leading to the improvement of RPV safety assessment of existing European LWRs under long-term operation and Generation III reactors under construction (supporting RPV ageing management and plant life extensions).
- Investigation of crack initiation by irradiation assisted stress corrosion cracking (IASCC).
- Investigation of growth (swelling) and creep under irradiation in internal structures by numerical methods for microstructure evolution and in situ verification of swelling macroscopic effects by in-plant measurements.
- Investigation of thermal ageing effect on the performance of the materials for long term operation.
- Investigation of galvanic corrosion on concrete reinforcement in the vicinity of the main cooling water pumps.
- Evaluation of chloride initiated corrosion of concrete reinforcements.
- Evaluation of polymeric material in concrete constructions and sealing applications.
- Investigation of polymer ageing mechanism, specifically irradiation ageing (influence of dose rate, synergy with vibrations, synergistic effects of irradiation, heat, moisture) and ageing affected by manufacturing.

4.3.2. Modelling of Ageing

The high variability of ageing and degradation mechanisms necessitates predictive tools to allow transferability and interoperability of the knowledge gained from limited experimental/empirical data (surveillance, in-field monitoring...). In this regard, one long term aim is to develop fully validated multiscale based models that link the nano scale through to the macro (i.e. structural) scale based on a multi-disciplinary approach. Better understanding of the physical mechanisms affecting the ageing of metallic materials combining advanced experimental investigations with the use of up-to-date modelling methods, such as those developed through the multiscale approach. Important is to investigate in depth the local phenomena and their interaction by using powerful numerical tools allowing an accurate prediction of ageing, not only of the components that are accessible for inspection, monitoring and repair but of all others. This extrapolation needs to be based on experimentally validated models using data from both in-pile and out of pile experiments.

One long term aim is to develop fully validated multiscale based models that link the nano scale through to the macro (i.e. structural) scale based on a multi-disciplinary approach

R&D topics

- Development/improvement of models to predict crack initiation and growth under various ageing conditions (environment, temperature, irradiation, loads ...).
- Simulation of welding, manufacturing processes and thermal ageing of critical components (pipes ...).
- Investigation of radiation induced loss of ductility and embrittlement, of reactor components by multiscale modelling taking into account the metallurgical variability of materials and their usage conditions.
- Development of computational fluid dynamics (CFD) based methodology for more accurate predictions of the operating conditions of each component.
- Investigation of the mechanisms influencing the dimensional stability of components (such as swelling, creep, fretting ...).
- Development of models to represent the effect of RAG ISA and other pathologies on concrete mechanical properties and civil work structure thermo-mechanical behaviour.
- Development of simulation methods to predict ageing in safety critical components and buildings.

4.4 Ageing Monitoring, Prevention and Mitigation

Component ageing needs to be monitored over the nominal and extended service life of SSCs in order to be able to correctly determine or anticipate the relevant ageing mechanisms and to evaluate the extent of degradation that may occur. The overall goal is to monitor and understand environmental conditions in the NPPs as well as their impact on the functionality of safety relevant components and structures in order to determine/predict/mitigate the effects limiting the lifetime of the safety relevant components. One way is to correlate the evolution of microstructure and material damage with applied loadings and environmental conditions. This will be particularly useful in the case of infrequent transients and will enable the operator to verify the suitability of maintenance programmes and in-service inspection, thus ensuring that operation remains within allowable limits.

The overall goal is to monitor and understand environmental conditions in NPPs as well as their impact on the functionality of safety relevant components and structures

4.4.1. Ageing Monitoring

The monitoring of ageing is largely in its infancy and much work is needed to satisfactorily meet the overall needs.

R&D topics

- Development of online monitoring tools for advanced water chemistries in BWRs and PWRs.
- Investigation of ultrasonic on-line monitoring in NPPs to improve methods for risk-informed inspection of piping and internals.
- Development of modelling capabilities to ensure the reliability of in-service monitoring tools.
- Development and provision of effective operational methodologies for inspection and tools for extended operation by consideration of such aspects as tightness of reactor containment, serviceability of waterways and the integrity of cooling towers.
- Development of methods for assessing the current permeability of concrete structures by sensing and monitoring the quality of the concrete.

- Development of non-destructive examination (NDE) methods for concrete (e.g. ultrasonic techniques).
- Detection of local and global degradation of cables and establishing a correlation for residual life estimation by using in-situ non-destructive electrical techniques for in-service full length measurements (for medium and low voltage cables).
- Development of monitoring techniques from the Motor Control Centre (MCC) providing remote assessment of the complete supply loop of electrical appliances.
- Sharing of best practices and the development of new techniques to detect ageing and to limit the degradation of I&C components and the evaluation of new technology and methods to be used for the purpose of I&C modernisation.
- Determination of a lifetime criterion for non-LOCA cables that are important for safety and also for non-safety critical cables.
- Determination of ageing monitoring methods for cables under the fire protection layer.

4.4.2. Prevention and Mitigation of Ageing

Prevention and mitigation of both ageing mechanisms themselves and their resulting damage and failure has been a long term challenge for engineers and scientists in many industries. However, this is an area where further studies and developments are required. The issue is associated with components that are usually very difficult and expensive to replace and may not be readily observable.

Prevention and mitigation of both ageing mechanisms themselves and their resulting damage and failure has been a long-term challenge for engineers and scientists in many industries

Many components are usually very difficult and expensive to replace, which requires the implementation of management and mitigation planning at the level of each NPP. However, collaborative R&D is badly needed both to maximise the efficiency of the different plans and to increase the reliability of system components.

R&D topics

- Reduction of radiation effects on an RPV including by fuel management and annealing.
- Optimisation of the secondary side water chemistry to minimise oxide deposition.
- Implementation of a condition information system supporting condition based maintenance (CBM) supported by worldwide experience and relevant data.
- Development of improved mitigation techniques to limit the occurrence of stress corrosion cracking: development of mitigation measures to be applied in the field including overlay welding, crack repair…
- Development of advanced primary water chemistry for VVER and PWR systems based on coolant treatment for radioactive waste reduction and lifetime extension of primary system components.
- Development of a corrosion protection of concrete reinforcements.

4.5 Functionality

4.5.1. Equipment Reliability

The reliability of each equipment is of critical importance even if it has no direct impact on the system safety as almost no component is operating in a stand-alone manner. Thus, the incremental development of each NPP to implement the improvement of the safety approaches necessitates a great R&D effort to establish if the systems are able to perform their intended function in a reliable and safe manner throughout the lifetime of their required use.

R&D topics

- Investigation of the reliability of relevant equipment for long term operation.

4.5.2. Industrial Obsolescence

NPPs are per definition designed to operate for a long time period during which a substantial technological and societal changes occur. Obsolescence essentially refers to components/systems that are no longer manufactured or maintained properly during the lifetime of an NPP. A common European approach should be developed with the help of NUGENIA either to create versatile technologies, possibly with other industries, or to adapt nuclear procedures to even faster evolving domains.

R&D topics

- Development of unified EU technical obsolescence management methods and procedures for components such as cables, electronics and feedthroughs.

4.5.2. Maintenance

R&D topics

- Justification of the monitoring systems supporting condition based maintenance (CBM).
- Integration of operational feedback (experience) provided by the monitoring systems into the planning of maintenance activities (CBM).
- Modernisation of pre-stressing control units to ensure fluent control of stress.
- Implementation of fleet wide monitoring data exchange systems in order to provide more accurate input data to maintenance teams.

4.6 Qualification

The qualification of methods used for the integrity assessment of SCCs is mandatory which necessitates the establishment of verification and validation of structural integrity assessment and lifetime procedures for SSCs and the development of specific materials or component tests. Often an overall qualification approach for structural integrity assessment and lifetime procedures requires a combination of advanced analytical and experimental qualification to comply with the functionality requirements for each component and system.

Qualification is inherent in performing R&D studies on the various issues relating to SSCs

R&D topics

- Development of specific and well controlled standard tests for specific material properties.
- Development of tests for the transferability of results from tests involving small specimen to structural components.
- Development of tests to assess material behaviour under specific degradation mechanisms which may require accelerated or modified experiments e.g. long term creep, irradiation effect, effect of transient (abnormal operation conditions).

TECHNICAL AREA 5 - Fuel Development, Waste and Spent Fuel Management and Decommissioning

i - Scope

Technical area 5 (TA5) covers development of nuclear fuel for existing, advanced and innovative core designs, aspects of fuel use in reactors (nuclear fuel behaviour mechanisms) and the fuel management steps – manipulation, transport and interim wet and dry storage. It also includes factors relating to the generation and management of radioactive waste, and the dismantling and decommissioning of nuclear power plants. It includes the safety issues linked with: fuel behaviour in normal operation and accident conditions, and the safety of the fuel cycle, in particular the investigation of criticality accidents and radioactive material dispersion.

Fuel behaviour in both normal operation and accident conditions currently is, and will continue to be, a major issue for the safe, secure and economic operation of nuclear power plants

TA5 has connections mainly with TA3 in parts of core optimisation and chemistry and to a lesser extent with TA6 regarding fuels for innovative LWRs and with TA1 for NPP safety and risk, in particular regarding criticality.

Outside of the scope of TA5 is radioactive waste disposal since IGD-TP is in charge of the research agenda and deployment plan for this topic.

The scope also takes account of emerging lessons from the Fukushima accident and proposes research, development and innovation to improve the safety and resilience of the existing and new build LWR reactor fleet.

ii - Objectives

The rationale of TA5 is to improve the safe, reliable and economic operation of Generation II and III NPPs (specifically in-reactor and out-of-reactor nuclear fuel management and radioactive waste management) and to maintain the sufficient level of safety defined by the regulatory bodies and reflecting the recommendations of the relevant international organisations.

Nuclear fuel production and use have reached a relatively mature state; nevertheless there is motivation to improve existing fuel types and to develop innovative fuel:

- to increase fuel safety margins and improve behaviour under operation and accident conditions
- to reduce reactor operating costs (including fuel costs)
- to reduce the amount and/or radio-toxicity of spent fuel
- to recycle existing waste (uranium, plutonium and minor actinides from prior reprocessing operations)
- to increase sustainability
- to improve the safety of fuel management and decommissioning/dismantling and more generally to the fuel cycle (such as criticality and re-suspension and transfer of radioactive materials)
- to improve proliferation resistance

The general R&D needs for all fuel types are:

- development of manufacturing techniques
- data on fuel and material properties
- post-irradiation examination (PIE) and collection of in-pile data on fuel performance (fuel thermo-mechanical and thermo-chemical behaviour under irradiation) and fuel resilience during accident conditions
- understanding and modelling of fuel performance and behaviour in accident conditions
- providing data for evaluating criticality risk in the fuel cycle, including the burn-up credit;
- source term and mitigation strategies in case of accidental dispersion of radioactive materials (re-suspension, filtering, …)

Despite the large knowledge base, there are still unknowns, necessitating dedicated fuel and material property, separate effect and semi-integral and integral testing to provide experimental data on fuel performance behaviour. These data can then be used to inform fuel development and to improve understanding and simulation of fuel performance.

An understanding of fuel behaviour in normal and abnormal conditions is underpinned by fuel R&D, which must address new design and safety requirements, increases in uranium enrichment, uranium and plutonium recycling (and potentially in the future minor actinide recycling), power up-ratings, and increased cycle length and burn-up. It must also address differences in behaviour engendered by incremental changes in the fuel components: that is, of the fuel pellets, cladding and assembly structural components.

Both spent nuclear fuel management and radioactive waste management have reached a relatively mature state, but there is still potential to be realised from the optimisation of management steps and the introduction of more efficient and reliable technologies leading to a reduction in cost and lower environmental impacts and also to improve the safety of processes.

The number and variety of nuclear facilities in the decommissioning stage will increase greatly in future years and therefore development of remote dismantling techniques and dose minimisation approaches are needed along with more reliable methods for the reuse and recycle of valuable materials and the release of other materials to the environment.

iii - State of the art

UO_2 enriched up to 5% in the form of solid or annular pellets in zirconium alloy cladding remains the most used fuel in European reactors. The main nuclear fuel suppliers in Europe are currently Areva, Westinghouse, GNF and TVEL (LWR fuels). MOX fuel is used in limited quantities mainly in France where processing is available. The existing expert and experimental base consists of vendors own R&D, and operational experience from utilities, research entities and international organisations (mainly IAEA, OECD NEA).

Burnable absorbers integrated into the fuel matrix (Gd and Er) are routinely used to control excess reactivity. Modifications to fuel microstructures have been recently introduced by incorporation of additives or by use of advanced manufacturing techniques. Fuel behaviour mechanisms are currently well known for UO_2 fuel in Zr cladding and AGR fuel for burn-ups up to 60 000 MWd/t. The fuel performance codes have been developed and validated (utilising data from operation and dedicated experimental programmes) and are routinely used for simulation of normal operation and accident scenarios.

Experimental facilities (research reactors, hot cells and laboratories) are available for research and testing.

Spent fuel management (with various nuclear fuel types for both commercial and research reactors) is undertaken and has benefited from the accumulated knowledge and experience of the past decades. Nevertheless, there is significant room for improvement and optimisation in various areas which would result in improved safety, security (proliferation resistance), and economic and environmental characteristics. The spent fuel management chain is carefully regulated by rules established by national regulators usually reflecting the recommendations of international organisations (IAEA). Within the EU a range of spent fuel storage arrangements are employed, in some countries fuel is stored primarily at the reactor stations/site where it is generated, whereas in other countries centralised storage is used for interim/long term storage after an initial cooling period at the reactor site. Transport of spent fuel is a well-established and regulated operation.

Fuel recycling of UO_2 and metallic fuels is generally well established within the EU, nevertheless development of knowledge is still required at process level

A number of decontamination, waste treatment and conditioning methods and technologies have been developed and are used. The management of special categories of waste has also been developed (tritium and C-14 waste, Be, graphite, mixed radioactive and chemically toxic waste, etc.). Nevertheless, the potential for improvement to reduce costs, risks and impacts is still far from exhausted. Methods of reuse and recycling of various materials (metals, concrete) have also been introduced. Experience from decommissioning and dismantling of nuclear facilities is being continually accumulated allowing drafting of guidelines and the use of best practices.

This knowledge is shared under the umbrella of international organisations (IAEA, OECD NEA – Working Party on Decommissioning and Dismantling).

iv - Challenges

- improving nuclear fuel allowing higher burn-ups and increased safety margins
- developing accident resistant fuels, innovative fuels and fuels allowing the burning of Pu and minor actinides (including non-UO_2 fuels)
- improving nuclear fuel reliability
- improving the quality of experimental data on fuel behaviour at high burn-ups and in accident conditions (through reduction of uncertainties) and extending data bases
- improving fuel performance simulation and computer codes, through an improved mechanistic science-based modelling and experimental validation
- maintaining of key experimental facilities (research reactors for irradiation, transient testing and safety-related experiments, hot cells and laboratories) and expanding their capabilities to meet future requirements
- handling and storage of leaking fuel assemblies (spent fuel pool and interim wet and dry storage) and handling of fuel and casks after longer term storage including the interface with a deep geological repository
- addressing the burn-up credit challenges (code validation and licensing issues)
- optimisation of spent nuclear fuel cycle and reprocessing and recycling of high burn-up and advanced fuels
- improving the safety of fuel management, of dismantling operations and of fuel cycle processes, regarding the risk of re-criticality and of dispersion and release of nuclear materials
- use of advanced IC tools for development of integrated waste management strategies
- minimisation of waste production due to design and material selection and operational measures and development of advanced waste treatment and conditioning technologies
- development of efficient dismantling technologies for structures and components, including remote dismantling techniques
- waste minimisation strategies for decommissioning, including safe release of material to the environment, recycle/reuse, disposal to VLLW repositories along with reliable and cost-effective activity measurement techniques

■ TECHNICAL AREA 6 - Innovative LWR design & technology

i - Scope & Objectives

The development of advanced Light Water Reactors (LWR) using innovative technology will allow assessing their possible key role for electricity production. These reactors could valuably make the bridge, throughout the 21st century, between the ageing nuclear installations currently in operation and/or the Advanced Generation III ones, now under construction, and the fourth generation reactors proposed by ESNII.

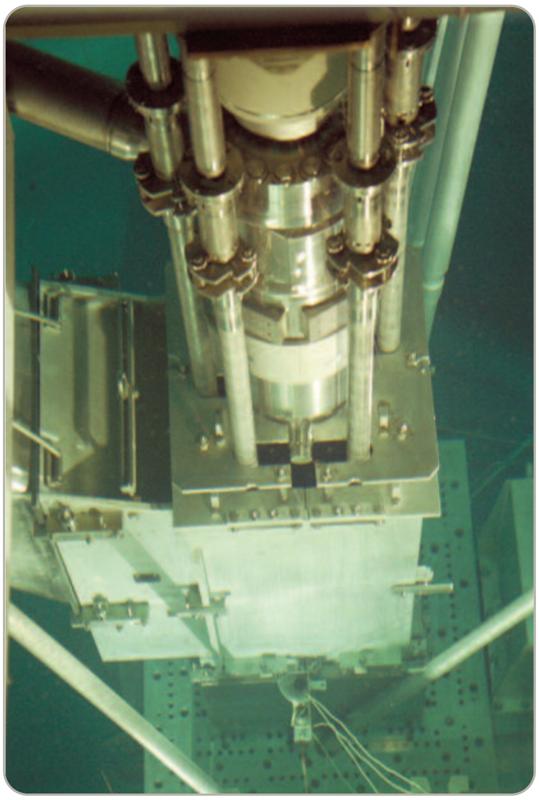

Figure 18: Phebus experience during operation (Cerenkov radiations) (Source: IRSN)

Technical Area 6 (TA6) provides guidance for setting up R&D projects on innovative technology to support the development of such innovative LWR designs.

Both existing and new LWR designs will profit from the expected R&D programmes in TA6 through the progress in the fields of safety & commissioning, operability, sustainability, economics and public acceptance.

The R&D proposed in TA6 - aimed at achieving long term operation, enhanced economics and safety by design for LWRs in operation and advanced ones, as well - will:

- contribute to improved sustainability with a better use of uranium resources and multirecycling capabilities of fissile materials
- be sized for smaller generated thermal power and modular construction techniques
- develop innovative component for reduced maintenance

Knowing that new technology deployment at the industrial scale could be a long time duration process, the following time lines will be considered:

- proposing evolutionary technology for mid-term application
- developing new LWR designs such as with higher conversion ratio or small modular reactors, expected to be ready for commercial operation by 15 to 20 years
- preparing breakthrough technology for a longer term future

Finally, the R&D proposed in TA6 should help to contribute to improve the public awareness and acceptance, which are mandatory to any new construction, especially after the Fukushima's events.

ii - State of the art

Synergies with all the other NUGENIA areas will be exploited for taking benefit from operating experience feedback from the current reactor fleet, prior to develop innovative technology that will result in LWR with improved performance.

That will include requirements for safety and performance, reduced maintenance and long term operation, material ageing related issues, control and repair, and plant power upgrade, as well.

The concept of a high conversion light water reactor has went-on being studied over the 80s in the aim at combining the advantages of LWR technology with the use of uranium – plutonium fuel, the achievement of high burn-up and optimised nuclear fuel consumption.

Small modular reactors are expected to have greater simplicity of design and reduced sitting cost compared to a current LWR. This results in a revival of interest worldwide.

iii - Challenges

The collaboration between industry, research organisations and universities should be fostered through the definition, implementation and launching of R&D projects. It is proposed to address the R&D needed work to support existing and new LWRs concepts, for achieving:

- Long Term Operation by design
- safety by design
- innovative component for reduced maintenance
- enhanced economics

Innovation, which should address basic technology, methods, testing and computation capacities, will be developed along with a transverse approach between all reactor concepts.

Indeed, screening new reactor models will foster and provide guidance for the development of new components and fabrication process, updated methods and calculation tool for assessing the design and safety performance. Conversely, having innovative solutions on shelves could make it possible to prepare the next generation of light water reactors.

All the necessary stages for reactor component design and fabrication should be addressed. Safety issues will be considered at the early stage of the design. Finally, the overall performance of the component for the related reactor concept will be assessed using new methods.

Those challenges will be addressed through the work done in the 5 sub-areas addressing:

- materials & innovative technology for reactor component design & construction
- innovative light water reactor concepts
- specific safety issues & approach
- key success factors for innovative LWR deployment
- public acceptance drivers for new build

6.1 Materials & innovative technology for reactor component design & construction

As a common theme for all LWR reactor concepts, advanced and breakthrough technologies for reactor component design and fabrication will be investigated. Synergies will be exploited to benefitting from the more demanding requirements and the progress made in basic technologies while anticipating new solutions for nuclear applications. Safety issues will be considered at the early stage of the design. Moreover, the overall performance of the component will be assessed adopting new methods.

R&D topics

- Investigation into innovative materials processing areas, surface engineering, nano-materials, composite materials and hybrids, to achieve properties suitable for high-performance nuclear components. Multiscale modelling will provide guidance for the evaluation of new alloyed materials.
- Investigation into component fabrication and welding processes, in line with the development of new materials including multi material assembly, complex geometry, near net shape fabrication, powder metallurgy processing. Specific development of non-destructive examination techniques is to be considered too.
- Analysis of specific issues related to innovative LWR design and performance assessment. It is worth achieving in-depth knowledge of local phenomena leading to component degradation mechanisms: corrosion, wear, fluid structure interaction, irradiation damage… Development of new testing capacities with high technology instrumentation, and computational methods will be two pillars for supporting this analysis.
- Contribution to the development of engineering simulator tools to assess the overall reactor system performance, including the environmental impact.
- Analysis of modular construction techniques.

6.2 Advanced LWR concepts such as: High Conversion ratio, Small modular reactor

Undertaking necessary R&D work is mandatory to the deployment of new LWR concepts within the next 15 to 20 years Fore exampleadvanced LWR with higher conversion ratio, longer fuel cycle, and capable of U-Pu multirecycling could help to identify potential improvements in the overall cycle sustainability.

Small modular reactors offer a flexible and progressive approach to nuclear capacity optimisation with limited infrastructure. These new concepts will foster and provide guidance for the development of innovative technology for reactor component design and fabrication, as described previously in sub-area1.

R&D topics

- Concept screening.
- Specific R&D topics derived from the previous screening: reactor physics, core cooling, compact component.
- Assessment of overall reactor system performance, in operating and accidental conditions.

6.3 Innovative LWR-specific safety issues & approach

The development of innovative materials, component fabrication processes, as well as the evaluation of new reactor concepts, could require specific safety approaches which should be integrated at the early stage of design. Harmonisation with EU Directives establishing a common framework for the nuclear safety of nuclear installations, likewise to WENRA statements and to IAEA safety publications is to be achieved.

R&D topics

- Exploitation of pre-normative research results to implement the safety requirements, including site selection and evaluation.
- Development of more sophisticated instrumentation and control systems for safety applications.
- Integration of the safety issues highlighted after the Fukushima Daiichi accident and by the EU stress tests specifications.

6.4 Key success factors for innovative LWR deployment

Key success factors for the deployment of innovative LWR reactors must be investigated vs. the deployment strategy of Generation IV systems and to the growing contribution from the renewable energy sources.

R&D topics

- Evaluation of flexible fuel cycle scenarios relying on a wide range of combinations of electricity sources for a postulated transition period, such as current and innovative LWR technologies, thorium cycle, Sodium fast reactors, and renewable energy sources. The outcome from ENEF analysis will be used as an input.
- Search for the enhanced operability of LWRs, including potential benefit for either load following or combined mode, and innovative technology for plant operation simplification.
- Evaluation of innovative solutions for minimising the environmental impact of LWRs.

6.5 Public acceptance drivers for new builds

In this sub area, it is proposed to address the rationales behind nuclear energy acceptance by the public, notably for new builds deployment. European policy is to provide with key elements for guidance.

R&D topics:

- Identification of the main drivers towards public acceptance for new build.
- Harmonisation of the communication policies, taking into account the differences in the public awareness and acceptance, which exist among the European countries.
- Organisation of the information dissemination.

ESNII fast reactor systems for sustainable fuel cycles

1. Introduction

One of the major concerns of society with regard to the implementation of nuclear energy is the high-level nuclear waste. Fast spectrum reactors with closed fuel cycle will allow a significant reduction of high-level nuclear waste radiotoxicity and volume. Fast reactors will also enable reduction in the use of natural fuel resources by a factor of around 50. In this way, it is clear that the use of fast reactors in a closed fuel cycle approach will allow a more sustainable implementation of nuclear energy.

The use of fast reactors in a closed fuel cycle approach will allow a more sustainable implementation of nuclear energy

For the development of these fast reactors within ESNII, it is of paramount importance to excel in safety, reliability, radiological protection and security:

- ESNII-systems shall be designed to reach at least the standards of safety, radiological protection and security put forward by WENRA for new reactors also targeting the Generation IV safety goals.
- ESNII-systems shall implement a safety approach based on the most recent standards and best international practices, using the experience gained from past and present nuclear science and engineering.
- ESNII-systems shall endeavour to reduce radioactive releases to the environment and doses to workers in normal and accidental situations to as low as reasonably achievable.
- Societal concerns in relation to nuclear safety and security shall be duly taken into account in the ESNII-systems design process.
- ESNII-systems shall be designed to have a high level of implementation of the concept of Defence-in-Depth.
- ESNII-systems shall be designed to use as much as possible passive safety systems and inherent safety characteristics.
- ESNII-systems will aim to practically eliminate the likelihood of severe accidents.
- ESNII-systems will be designed to be robust against the Fukushima accident initiators.
- Security shall form an integral part of the ESNII-systems design.
- ESNII-systems shall aim at further improving the economic competiveness and operability of nuclear energy in a future European energy mix.

The main objective of ESNII is to keep European leadership in fast spectrum reactor technologies that will excel in safety and will permit a more sustainable development of nuclear energy. With respect to the 2010 evaluation of technologies, sodium is still considered as the reference technology, since it has broader technological and reactor operations feed-back. The lead-bismuth fast reactor technology has significantly extended its technological base and can be considered as the shorter-term alternative technology, whereas the gas fast reactor technology has to be considered as a longer-term alternative option. The main goal of ESNII is to design, license, construct, commission and put into operation before 2025 the sodium fast reactor prototype reactor, called ASTRID and the flexible fast spectrum irradiation facility MYRRHA.

The main goal of ESNII is to design, license, construct, commission and put into operation, before 2025, ASTRID and MYRRHA

ASTRID will allow Europe to demonstrate its capability to master the mature sodium technology with improved safety characteristics responding to society's concern of having the highest level of safety possible. Therefore, the design of ASTRID focuses on meeting the challenges in terms of industrial performance and availability, improved waste management and resource utilisation and a safety level compatible with WENRA objectives for new nuclear builds and aiming at achieving the Generation IV goals. An associated R&D programme will continue to accompany and support the development of ASTRID to increase the lines of defence and robustness of the safety demonstration of this technology, and allow the goals of the Generation IV to be reached, not only on safety and proliferation resistance, but also on economy and sustainability.

With MYRRHA, Europe will again possess a flexible fast spectrum irradiation facility that is a prerequisite for further innovations in fuels, materials and components for fast spectrum reactors. It will further diversify Europe's expertise in fast reactor technology and allow major innovations towards even more economic, sustainable and safe reactor concepts. Since MYRRHA will be conceived as an Accelerator Driven System, it will be able to demonstrate this technology, thereby allowing the technical feasibility of one of the key components in the double strata strategy for high-level waste transmutation to be evaluated. Due to the fact that MYRRHA will be based on heavy liquid metal technology (namely lead-bismuth eutectic), it can serve the role of Lead Fast Reactor European Technology Pilot Plant (ETPP) as identified in the LFR roadmap. An associated R&D programme will accompany and support the development of MYRRHA.

For the financing of the total investment cost of these facilities, it will be of paramount importance to establish the appropriate consortium structure and legal basis, allowing candidate consortium members to identify the added value of the facility for their own interest.

In parallel to the realisation of ASTRID and MYRRHA, activities directed towards the Lead Fast Reactor technology and the Gas Fast Reactor technology should be continued taking into account their specific needs.

For the development of the Lead-cooled Fast Reactor, maximum synergy of activities will be sought with the MYRRHA development to optimise resources and planning. For the LFR demonstrator ALFRED, the main focus should be on design activities typical for a critical power reactor connected to the grid, as well as on R&D activities on the lead coolant, addressing the specific characteristics that differ from lead bismuth. Design activities and support R&D shall be performed in the coming years to the maximum extent compatible with available resources and taking full advantage from feedback, where applicable, from the ongoing design of MYRRHA and related R&D programmes. These activities will allow the LFR consortium to reach the level of maturity needed to start the licensing phase and then the construction of ALFRED, provided that adequate financial resources are made available.

For the development of ALFRED, the Lead Fast Reactor demonstrator, maximum synergy of activities will be sought with the MYRRHA development

In addition to the closure of the nuclear fuel cycle in a sustainable manner, the Gas Fast Reactor has the potential to deliver high temperature heat at 800°C for chemical process, production of hydrogen, synthetic fuels, etc. The Helium-cooled Fast Reactor is an innovative nuclear system having attractive features: helium is transparent to neutrons and is not chemically reactive. Its viability is however essentially based on two main challenges. First, the development and qualification of an innovative fuel type that can withstand the irradiation, temperature and pressure conditions put forward for the GFR concept. Secondly, a high intrinsic safety level will need to be demonstrated for this GFR concept. This will imply dedicated design activities followed probably by out-of-pile demonstration experiments. These high priority R&D activities should be embedded into an overall R&D roadmap in support of the development of the Gas Fast Reactor concept. For the development, guidance and implementation of this R&D effort, a GFR centre of excellence will be created. It might open up the technical capability to launch the ALLEGRO gas cooled demonstrator.

Based on the ADRIANA project, a number of supporting facilities for the different systems and technologies have been identified, in addition to experimental reactors like JHR and PALLAS. The realisation and operation of these supporting facilities, in particular a fast reactor MOX production line, will be of primary importance to reach the aforementioned objectives. Maximum synergies between the different Liquid Metal Reactor technologies will be exploited, for instance in the field of instrumentation and thermal-hydraulics.

Raising the financial resources to deliver the ESNII projects and build the different facilities will be a key factor of success. In this respect, international collaboration through GIF and bilateral or multi-lateral frameworks will be looked for to optimise resources.

■ The ASTRID Project (Advanced Sodium Technological Reactor for Industrial Demonstration)

Objectives

The objective of the integrated technology demonstrator ASTRID is to ensure industrial-scale demonstration of a Generation IV Sodium Fast Reactor, meeting the highest level of safety and security standards, and providing significant improvements in terms of industrial operation. The reactor is expected to operate around 2020.

The ASTRID programme encompasses the ASTRID reactor itself, the realisation of sodium technological loops and the validation of components, as well as the construction of a fuel manufacturing workshop.

Benefiting from the accumulated operation of more than 400 reactor-years in SFR technology, the key objective of ASTRID is to demonstrate at industrial scale significant improvements to meet Generation IV standards, qualifying innovative options in well-defined areas (safety and operability), while providing a test bench for advanced in-service inspection and repair techniques. ASTRID will also have provisions for experiments on transmutation of minor actinides (mainly americium) in significant quantities to allow optimised management of wastes.

The ASTRID safety options will be compliant with the highest safety standards, including lessons learnt from the 9/11 events and the Fukushima accident.

Though future fast reactor plants intend to be breeders, ASTRID will be an iso-generator keeping in mind the current nuclear material situation.

With the associated closed fuel cycle, ASTRID will meet the preservation of resources priority, allowing the optimisation of uranium resources as well as the multirecycling of plutonium, and the reduction of the quantity, the half-life and the toxicity of ultimate waste (minor actinides), while providing a low-carbon, intensive energy source.

It will be equipped for experiments. Its design must therefore be flexible enough to be able to test innovative options that were not chosen for the initial design. Novel instrumentation technologies, new fuels and even new system components will be tested in ASTRID.

ASTRID will be available for irradiation experiments like those conducted in Phénix in the past. These experiments will help to improve the performance of the core and absorbers, as well as to test new fuels and structural materials, such as carbide fuel and oxide dispersion steel (ODS) cladding. ASTRID will be equipped with a hot cell for examining irradiation objects, built either in the plant or nearby.

ASTRID aims at ensuring the industrial demonstration of a Generation IV Sodium Fast Reactor

The ASTRID industrial prototype, with the main objective of confirming long term innovative options at larger scale, both for the development of the Generation IV Sodium Fast Reactor, but also for the fuel cycle and waste management, will represent a key component in the development of future Generation IV nuclear systems.

State of the art

Following French Government orientations given on the sustainable management programme for radioactive materials, stipulating the commissioning of a Generation IV reactor by 2020, the ASTRID programme has been launched by the CEA, gathering European and international public and private partnerships. It is based on decades of continuous R&D work on sodium reactor technology, some of which has been performed through European cooperation in projects such as CP-ESFR and EISOFAR (realisation of a sound scientific and technical basis for the European sodium fast reactor), THINS (Thermal-hydraulics of Innovative Nuclear Systems), FREYA (Fast Reactor Experiments for hYbrid Applications), SARGEN IV (Safety Assessment for Reactors of GENeration IV), MATTER (MATerials TEsting and Rules), GETMAT (Generation IV and transmutation materials) ANDES (Accurate Nuclear Data for Nuclear Energy Sustainability), and HELIMNET (HEavy Liquid Metal NETwork).

The ASTRID project management comprises a project team responsible for the industrial architecture, managing most of the design work of ASTRID that is performed by several industrial partners. These industrial partners have entered collaboration with CEA on ASTRID design studies and are currently contributing on their own budget through a cost sharing scheme. The ASTRID project team is supported by CEA R&D departments through a dedicated set of projects delivering R&D results following expression of needs by the ASTRID project.

Since September 2012, the following partners have been involved in the conceptual design phase:

- AREVA NP: nuclear island (core and fuel stays with CEA)
- EDF: support to the owner and contribution to R&D
- ALSTOM: turbine island
- BOUYGUES for civil engineering
- COMEX NUCLEAIRE: innovative studies in robotics and mechanics
- TOSHIBA for development of large electromagnetic pumps
- JACOBS for infrastructures
- ROLLS-ROYCE for research and technology development on sodium gas exchangers and fuel handling
- ASTRIUM for reliability, maintainability and availability analysis

And other partnerships under discussions.

The conceptual design phase will result from the assessment of technical options and safety orientations, focusing on innovations concerning the design of the core, the decay heat removal systems, the core-catcher and elimination of sodium-water reactions.

ASTRID will be a pool type, sodium cooled fast reactor of 1500 MWth, generating about 600 MWe. That level of power is required to guarantee representativeness in terms of design, operation and safety demonstration, of the reactor core and main components, and will compensate for the operational costs by generating a significant amount of electricity.

The ASTRID design benefits from the advantages of pool-type, sodium-cooled fast neutron reactors, which provide very favourable inherent safety margins with regard to Fukushima-like events:

- The main vessel contains the whole primary system including the core, the intermediate heat exchangers and the primary pumps, giving pool-type SFR a high level resistance to loss of coolant accidents (LOCA), since it is possible to install a guard vessel around the main vessel. Furthermore, the primary system is not pressurised.
- The intermediate system (or secondary system) uses sodium loops to transfer the energy from the primary circuit to the main heat exchangers and provides an additional barrier.
- The size and mass of the primary system, along with the quantity of primary coolant and its physical properties provides for a very large thermal inertia of the reactor, the thermal inertia allowing larger grace times in order to put in operation the DHR systems.

Design and construction rules for mechanical components such as vessel, piping, support structures, are already provided for by available codes and standards. ASTRID uses mainly the RCC-MRx code developed especially for Sodium Fast Reactors, Research Reactors (like JHR) and Fusion Devices (like ITER), which take into account the past experience on RCC-MR and operational feedback from Phénix and SuperPhénix.

For structures, materials with good feedback operation on Rapsodie, Phénix and SuperPhénix have been selected. After Phénix's definitive shutdown in 2009, a dedicated programme of structural material examination has been set up for the coming years in order to improve current material databases.

The ASTRID core is an heterogeneous MOX core called CFV ("Cœur à Faible effet de Vide sodium" or "Low sodium Void Worth Core"), that is characterised by a sodium void coefficient close to 0. This is a major difference from classical fast neutron reactor designs, and provides the ASTRID core with improved inherent behaviour in terms of safety.

ASTRID technology benefits also from the large fuel qualification database obtained from former reactors. The ASTRID start-up core will be based on MOX fuel with 15/15Ti AIM1 alloy cladding that was irradiated in the Phénix reactor and showed good performance. After Phénix's definitive shutdown in 2009, a dedicated programme of post-irradiation examination has been developed for the coming years so that the most up-to-date knowledge on AIM1 cladding will be available. This examination programme will also give feedback on the heterogeneous core concept. The material for the hexagonal tubes will be EM10, used in the last Phénix core batches and well qualified.

To ensure waste reduction capability, ASTRID will continue the demonstration at higher scales of minor actinide transmutation

To ensure waste reduction capability, ASTRID will continue the demonstration at higher scales of minor actinide transmutation (mainly americium) that was started at experimental scale with Phénix.

As the prototype of Sodium Fast Reactor technology, ASTRID has the main objective of demonstrating advances on an industrial scale, by qualifying innovative options in order to meet the requirements of future electricity-producing reactors in the following areas:

- core meltdown probability at the lowest achievable, including the fulfilment of 2010 WENRA[18] objectives for new nuclear power plants
- minimisation of potential mechanical energy release in case of core degradation
- improvement of the means for inspecting the structures in sodium
- reduction of risks associated with the affinity between sodium and oxygen
- reduction in the duration of programmed outages (fuel handling & maintenance) and unforeseen shutdowns

Challenges

Meeting Generation IV standards for SFR

Core meltdown probability at the lowest achievable

With the objective of reducing the probability of core meltdown and/or limiting energy release accidents potential, core options are being studied: the CFV (Cœur à Faible effet de Vide sodium) core concept based on a low sodium void effect involves a heterogeneous axial $UPuO_2$ fuel with a thick fertile plate in the inner core and characterised by an asymmetrical, crucible-shaped core with a sodium plenum above the fissile area.

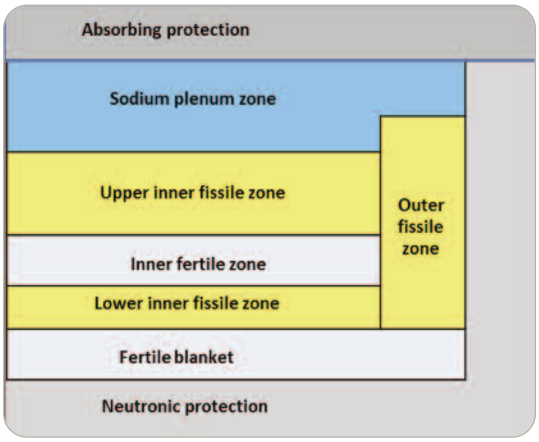

Figure 19: CFV Core (Source: ASTRID Consortium)

Initially conceived with the intention of significantly reducing the sodium void effect in case of sodium boiling compared with the SFRv2 concept, the CFV core concept focuses on optimising the core neutron feedback parameters (reactivity coefficients) so as to obtain improved natural core behaviour during accidental conditions leading to overall core heating. More specifically, the reactivity effect associated with sodium expansion achieved by design (sodium plenum and heterogeneous fertile plate) is negative in the event of a total loss of primary coolant, and can result in an overall negative void effect if a boiling phase is reached.

This innovative specificity in comparison to standard Superphénix or EFR cores can be extrapolated and remains valid for high-power CFV cores. The preliminary studies on unprotected-loss-of-flow (ULOF) transients or unprotected-loss-of-heat-sink (ULOHS) transients show potential for an acceptable natural behaviour of the CFV core.

The CFV concept also shows a low reactivity loss per cycle thanks to the large diameter fuel pins. This geometry leads also to longer cycles and fuel residence times, as well as improved behaviour during an accidental control rod ejection transient (no ejection in SFR but inadvertent control rod withdrawal) with respect to conventional core designs.

These characteristics of the CFV core concept will remain to be confirmed by future simulation and experimental validation.

18 - Western European Nuclear Regulators' Association - Statement on « Safety objectives for new nuclear power plants » (November 2010)

ASTRID will be equipped with additional safety device to enhance the robustness of some safety margins. One example is a passive-type emergency shutdown system patented by the CEA, which is called SEPIA. However further R&D will be devoted to analyse alternative systems, including some ideas proposed during the EFR project.

Significant R&D effort is ongoing and will be increased to improve instrumentation and measurement systems to reinforce core and reactor monitoring.

Prevention will be consolidated by making sure all components important for safety can be inspected, as well as components capable of impacting these safety-important components. This first and foremost concerns the internal structures of the reactor block, particularly the core support and core cover plug for which efficient inspection methods must be qualified. The choice between the different reactor block internal structures described further takes into account this inspection criterion.

ASTRID will be equipped with a third passive-type emergency shutdown system

Decay heat removal (DHR) systems will be sufficiently redundant and diversified so that the practical elimination of their total failure during a given duration can be demonstrated. Development of dedicated DHR systems using structures for degraded situations is to be strengthened in order to have a diversified DHR system, as usual systems pass through the roof slab; such a design option needs to be studied in relation to the concept of a core catcher external to the primary vessel (see following picture).

Resistance to a potential mechanical energy release accident

For the safety demonstration, in particular prevention, a core catcher will be installed in ASTRID and will be designed to recover the entire core, maintain the corium in a sub-critical state while ensuring its long-term cooling, as well as being inspectable. Several options need to be investigated as to possible core-catcher technologies, locations (in-vessel or outside the vessel) and performances.

In compliance with the WENRA approach on the independence of lines of defence, the containment will be designed to resist mechanical energy release with the objective that

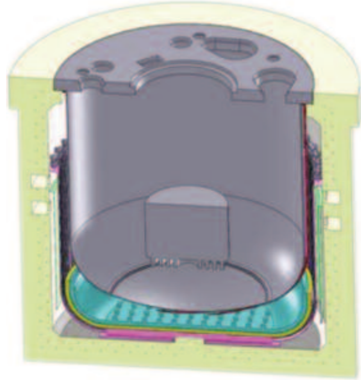

Figure 20: Example of a intervessel core catcher with innovative DHR by structures (CEA patent) (Source: ASTRID Consortium)

no countermeasures are necessary outside the site boundary in the event of an accident. R&D will be needed to support the demonstration.

Inspecting structures in sodium

Contrary to the Phénix and Superphénix reactors, periodic inspection of the reactor block internal structures has been integrated into the design. Although some technologies now exist that enable this inspection either from outside or inside the vessel, further R&D on optical and ultrasonic systems will be necessary to develop and select the most suitable technology to be used in the primary system.

After a comparison of innovative pool designs, a reference design has been selected from the results of a multi-criteria analysis: assessment of robustness (life expectancy, thermo mechanical behaviour…), global In-Service Inspection and repair possibilities, and economic factors. The selected pool type is with a conical redan, an improvement on the EFR project.

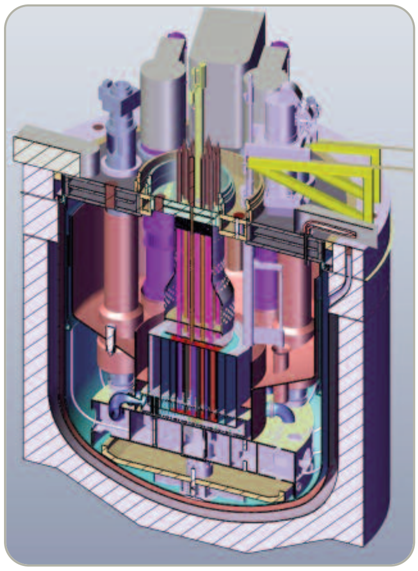

Figure 21: Design primary system (Source: ASTRID Consortium)

Reduction of risks associated with the affinity between sodium and oxygen

To improve the safety and acceptability of the reactor with the de facto elimination of risks associated with sodium-water reactions, an innovative energy conversion system is being considered that uses gas for the thermodynamic transformations (Brayton cycle). This type of system has been studied by CEA in the past and has been adapted to the pressure and power ranges required in ASTRID. Further work is needed to couple this concept to the reactor through an intermediate sodium system, in order to exclude any risk of gas entrainment into the core.

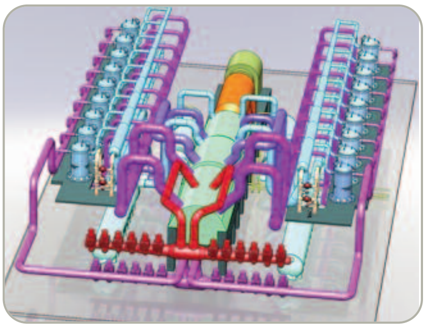

Figure 22: Example of layout with a gas energy conversion system (Source: ASTRID Consortium)

In case the water-steam cycle is to be retained, further improvements through R&D are to be considered on:

- Modular steam generators, whose size guarantees the integrity of the intermediate heat exchanger, and thus protects the primary system, the secondary system and the steam generator casing, even in the event of the sudden and simultaneous failure of all the steam generator module tubes.
- Steam generator concepts that ensure better protection against propagation of tube failure in case of sodium-water reaction.

The redundancy and performance of the leak detection systems will also be reinforced.

Complementary studies are needed for the improvement of the efficiency and reliability of the systems designed to detect sodium leaks and fires, as well as the possibility of installing double-envelope pipes, inerting the rooms or using confinement measures to stifle sodium fires.

Reduction in the duration of programmed outages (fuel handling & maintenance) and unforeseen shutdowns

Innovative options have been identified to improve fuel handling system performance and reliability; taking into account transmutation fuels and their cooling times, fuel loading and unloading will continue to be performed in sodium since this provides greater operating flexibility in normal and accident conditions.

A specific methodology will be applied to the overall reactor design to reduce the causes of unavailability by focusing on the following design aspects: reliability of equipment tested using proven technologies, simple and robust equipment designs, and preventive detection of failures.

In order to reduce unavailability times, research on removability of equipment, with in situ maintenance and handling studies to optimise all handling, cleaning, repair and requalification operations from the design phase, need to be performed.

For the protection of investments, the project will focus on making as many reactor structures as possible reparable (or replaceable); although designed for 60 years, the core cover plug – a component sensitive to thermo mechanical load and mechanical hazards – will have to be potentially replaceable.

The MYRRHA Project (Multipurpose Hybrid Research Reactor for High-tech Applications)

Objectives

The first objective of MYRRHA is to establish a multipurpose research facility serving as a flexible fast spectrum irradiation tool in support of technology development (in particular for materials, components and fuel irradiation tests) of the three fast reactor systems (SFR, LFR and GFR). Also, MYRRHA will offer a wide range of interesting irradiation conditions for fusion material research. As a multipurpose research facility MYRRHA was included in the high priority list of ESFRI. MYRRHA will be conceived as an Accelerator Driven System, able to work in critical and subcritical mode.

The combination of Partitioning and Transmutation (P&T) and dedicated burner technologies such as ADS is proposed in order to relax constraints on geological disposal.

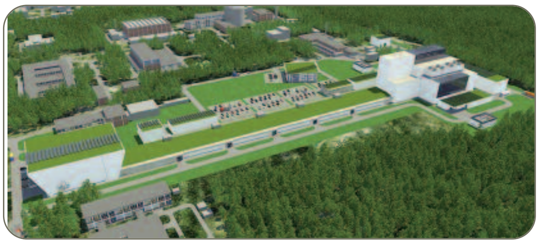

Figure 23: MYRRHA Layout picture
(Source: MYRRHA Consortium)

Hence, since ADS represents a possible major component in the P&T framework, the demonstration of the sub-critical dedicated burner option is needed. The MYRRHA project proposed by SCK•CEN responds to this need. The main objectives of MYRRHA are the demonstration of the ADS concept at a reasonable power level on the one hand and, on the other hand, the proof of technical feasibility of transmutation of minor actinides…

MYRRHA will play the role of European technology pilot plant (ETPP) in the roadmap for LFR

As an ADS, MYRRHA contains a proton accelerator of 600 MeV, 4 mA, a spallation target and a multiplying core with MOX fuel, cooled by liquid Lead-Bismuth Eutectic (LBE).

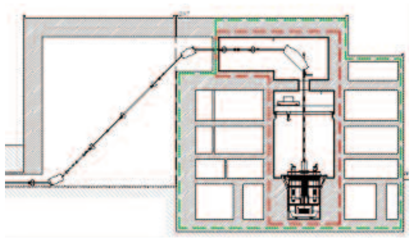

Figure 24: MYRRHA building vertical cut with beam line entrance picture to be added
(Source: MYRRHA Consortium)

Since MYRRHA is based on heavy liquid metal technology, it will strongly contribute to the development of lead fast reactor (LFR) technology. MYRRHA will play the role of European technology pilot plant (ETPP) in the roadmap for LFR.

State of the art

Following the decision of the Belgian Government to support the MYRRHA-project, SCK-CEN has set up a project structure and team integrating the design and R&D efforts expended in several Seventh Framework Programme (FP7) projects like the CDT (Central Design Team), SEARCH (Safe ExploitAtion Related CHemistry for HLM reactors), MAX (MYRRHA Accelerator eXperiment), MAXSIMA (Methodology, Analysis and eXperiments for the "Safety In MYRRHA Assessment), THINS (Thermal-hydraulics of Innovative Nuclear Systems), FREYA (Fast Reactor Experiments for hYbrid Applications), LEADER (Lead-cooled European Advanced DEmonstration Reactor), SARGEN IV (Safety Assessment for Reactors of GENeration IV), SILER (Seismic-Initiated events risk mitigation in LEad-cooled Reactors), MATTER (MATerials TEsting and Rules), GETMAT (Generation IV and transmutation materials), ANDES (Accurate Nuclear Data for Nuclear Energy Sustainability), and HELIMNET (HEavy Liquid Metal NETwork).

During the 2010-2014 FEED (Front-End Engineering Design) period the following items will be accomplished:

- primary system and plant design and the associated R&D programme
- pre-licensing process
- set-up of the international consortium

SCK-CEN will be responsible for the primary system, but all other systems, structures and components together with the plant layout will be subcontracted to an international industrial consortium (called FEED-engineer) by public tendering.

For the design of MYRRHA, as much as possible benefit has been taken from previous fast reactor programmes to relax the licensing process. The objective of MYRRHA is also to excel in safety by practically eliminating Fukushima-accident initiators by means of redundant and diversified fully passive decay heat removal systems. Special attention will also be given to design choices and measures for prevention and hence practical elimination of severe accident scenarios.

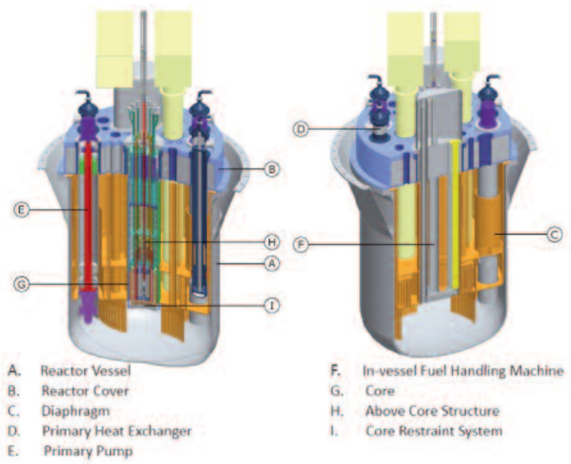

Figure 25: MYRRHA reactor (Source: MYRRHA Consortium)

MYRRHA is a pool-type ADS having its primary and secondary systems designed to evacuate a maximum core power of 100 MWth.

MYRRHA is a pool-type ADS having its primary and secondary systems designed to evacuate a maximum core power of 100 MWth

All the MYRRHA components are optimised for the extensive use of the remote handling system during component replacements, inspection and handling. Figure 25 shows a section of the MYRRHA reactor revealing its main internal components.

The accelerator is the driver of MYRRHA since it provides the high energy protons that are used in the spallation target to create neutrons which in turn feed the subcritical core. The accelerator will provide a proton beam with an energy of 600 MeV and an average beam current of 4 mA. High availability is expressed by a long Mean Time Between Failure (MTBF) and is commonly obtained by a combination of over-design and redundancy. On top of these two strategies, a fault tolerance scheme will be implemented to allow the accelerator to recover the beam after failure within a beam trip duration tolerance of 3 s.

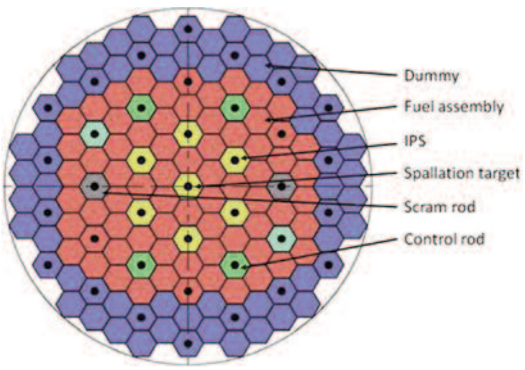

Figure 26: Cut in the MYRRHA/FASTEF core, showing the central target, the different types of fuel assembliesand dummy components.
(Source: MYRRHA Consortium)

The reactor core (previous figure) consists of mixed oxide (MOX) fuel pins, typical for fast reactors. The requested high fast flux has been obtained by optimising the core configuration geometry (fuel rod diameter and pitch) and maximising the power density. The use of lead-bismuth eutectic (LBE) as coolant permits lowering of the core inlet operating temperature (down to 270°C) decreasing the risk of corrosion and allowing the increase core ΔT.

In subcritical mode the spallation target assembly, located in the central position of the core, brings the proton beam via the beam tube into the central core region. The assembly evacuates the spallation heat deposited there, presents a barrier between the LBE and the reactor hall and assures optimal conditions for the spallation reaction. The assembly is conceived as an In-Pile-Section (IPS) and is easily removable and replaceable.

The encumbrance of the core with the proton beam, the fact that the space situated directly above the core will be occupied by lots of instrumentation and IPS penetrations and the core compactness result in insufficient space for fuel handling to load/unload the core from above. Since the very first design of MYRRHA, fuel handling is thus performed from underneath the core.

The major technological issues for the MYRRHA demonstrator are:

- lead-bismuth chemistry control and conditioning
- lead-bismuth component testing and thermo-hydraulics
- lead-bismuth instrumentation
- material qualification
- driver fuel qualification
- coupling technology of accelerator with subcritical core
- high intensity proton accelerator performances and reliability

Challenges

Lead-Bismuth chemistry control and conditioning

For long-term operation of a LBE cooled ADS, chemistry monitoring and control are crucial for the reactor. A LBE chemistry control and conditioning R&D programme involves the technology related to chemical control of the coolant and purification of the evaporated elements that have low retention in LBE such as Hg. Several issues have been identified for this programme: the development of oxygen sensors to measure the dissolved oxygen concentration in the coolant, the conditioning of the LBE to minimise dissolution of structural materials and core internals and to prevent formation and precipitation of oxides, filtration and trapping of impurities in the LBE, the evaporation and capture of volatile and/or highly radiotoxic elements (e.g. Po-210) from the cover gas and finally the removal of LBE or dissolved constituents from among other components and test samples.

Lead-bismuth component testing and thermal-hydraulics

In order to secure safe and reliable operations of MYRRHA, an extensive R&D programme is set up to develop and test reactor components. Since MYRRHA is an experimental reactor, fuel handling is a rather frequently occurring task. Two fuel handling machines will be used, based on the rotating plug concept. A high level of reliability of these machines is crucial. Further (thermal-hydraulic) analysis of the fuel assemblies, the MYRRHA core (including control and safety rods), the spallation target and reactor pool is needed to assure long-term operation of the liquid metal cooled MYRRHA reactor. Within the research and qualification programme of the LBE components, the proper working of the primary heat exchangers and the primary pumps must be confirmed.

In-service inspection in Lead-bismuth

The use of LBE as coolant in MYRRHA has also some known disadvantages: its opacity complicates maintenance and fuel handling operations. Developments regarding ultrasonic techniques must be made to improve these operations.

Material qualification

Based on available data on mechanical and thermal properties, irradiation performance, manufacturing and availability, the following steels have been selected as the candidate materials for the components:

- titanium stabilised austenitic stainless steel 15-15Ti for e.g. the fuel cladding of the first cores
- ferritic-martensitic steel T91 for e.g. the spallation target window
- austenitic stainless steel type 316L solution annealed for the majority of other components including the reactor vessel, the heat exchangers, the diaphragm and the core barrel

The austenitic stainless steels including 316L and 15-15Ti have been extensively used in construction of fast sodium cooled reactors in Europe, US and Japan and are therefore relatively well characterised for nuclear applications. However, the innovative nature of the MYRRHA installation poses new challenges for material performance, particularly because of the lead-bismuth eutectic coolant which could be quite corrosive under certain conditions and also might affect the mechanical properties.

The efforts are distributed over the following five overlapping activities:

- identification of key material issues for design and licensing of MYRRHA
- development of test and evaluation guidelines for characterisation of structural materials
- assessment of material properties
- development of testing infrastructure
- qualification of the chosen materials for the MYRRHA conditions

Fuel qualification

At present stage of definition of the fuel R&D programme, attention is mainly paid to the driver fuel and cladding material. MYRRHA will rely on conventional fast reactor MOX fuel technology developed and demonstrated in previous sodium programmes like SNR-300, (fuel licensing process only) RAPSODIE, Phénix and Superphénix. The feedback from these programmes covers in many aspects the operating conditions of MYRRHA fuel. Return of experience is maximised and licensing needs are minimised further by choosing the Phénix fuel pin design and cladding material (15-15 Ti) as the preferred option. However, since MYRRHA will have LBE as coolant the topics of clad-coolant interaction and fuel-coolant compatibility have not been dealt with in sodium programmes and are embedded more extensively in the MYRRHA R&D programme.

Innovative MA bearing fuels will be loaded in MYRRHA to allow for a further screening and down-selection of these new types of fuels and to finally allow qualification of these innovative fuel types.

Coupling technology of the accelerator with subcritical core

In an Accelerator Driven System the coupling of the accelerator, the target and the subcritical core deserves special attention. The reactor physics of such a coupled system is significantly different from a critical system and dedicated experiments are needed. More specifically, the accurate on-line monitoring of the subcriticality level needs to be validated. To respond to this question, the accelerator-coupled GUINEVERE (Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor) experiment was conceived and validation experiments will need to be carried out.

For the coupling of the accelerator and the subcritical core, the beam window serves as a barrier. A dedicated programme on the qualification of the beam window under different MYRRHA conditions is being carried out. Based on the feedback from the MEGAPIE-experiment and the MYRRHA operating characteristics for the beam window, the licensing approach will be based on a maximum supposed lifetime of the beam window of one cycle (3 months operation). In the beginning, the window will be replaced after every cycle while a qualification programme will run in parallel during the first cycles to demonstrate the longer lifetime of the beam window.

High intensity proton accelerator performances and reliability

For MYRRHA a 600 MeV linear proton accelerator with a nominal design current of 4 mA is envisaged. Linear accelerators of this type have been constructed in the past. However, the reliability requirement for the MYRRHA Linear Accelerator is more than one order of magnitude more stringent than what is commonly achieved in research accelerators. Preliminary analyses have shown that in principle the required reliability level should be feasible. However, the realisation of the goal makes a research programme on the accelerator indispensable.

The accelerator R&D programme is focussing on:

- injector developments @ UCL
- main Linac component developments
- global accelerator design
- system optimisation

In collaboration with UCL (Université Catholique de Louvain), the first part of the MYRRHA accelerator will be built and tested: injector developments @UCL. This first part, consisting of an ion source, low energy beam transfer line, a 4-rod based Radio-Frequency Quadrupole (RFQ) and a diagnostic section, will deliver protons of 1.5 MeV. The aim is to test first the MYRRHA injector section to analyse and if needed improve its reliability in view of the overall reliability targets.

To allow the implementation of a fault-tolerance capability, which is of crucial importance for reliability enhancement, prototypes of the different Linac components will need to be constructed. It is planned to carry out reliability tests by means of prototypes of each cryomodule 'family':

- the SuperConducting Cross-bar H-type (SC-CH) cryomodule
- the spoke cryomodule
- the long elliptical cryomodule

Besides the cryomodules themselves, critical components to be developed are:

- the superconducting Radio Frequency (RF) power couplers
- the Low Level RF (LLRF) required for the fault tolerant scheme

The Linac design will be consolidated by means especially of advanced beam simulations based on start-to-end simulations and associated error analyses together with assessments on new R&D results, new reliability studies, definition of preliminary infrastructures and revised cost estimates. Recommendations will be formulated, including a roadmap towards the actual construction of the MYRRHA accelerator. Links between the activities of the FP7 project MAX, the results of the FP7 project CDT, and the related R&D ongoing in the accelerator community will be set-up.

During the FP6 EUROTRANS project (European Research Programme for the Transmutation of High Level Nuclear Waste in an Accelerator Driven System), a preliminary reliability study of the ADS reference accelerator has been conducted in order to assess the number of beam trips. Such beam trips threaten the core materials and can affect dramatically the plant availability. It is intended to pursue these reliability-oriented studies and to develop a more accurate reliability model of the MYRRHA accelerator. A model of the full MYRRHA Linac will be built taking into account all support systems and, as far as possible, smart control strategies, fast beam shutdown systems and accelerator/reactor interface aspects.

The ALFRED Project (Advanced Lead Fast Reactor European Demonstrator)

ALFRED is the Advanced Lead Fast Reactor European Demonstrator whose conceptual design has been carried out as part of the 7th FP LEADER project. The work capitalises on achievements of previous FWP projects on heavy liquid metal cooled fast reactor technologies, such as ELSY, GETMAT, and EUROTRANS. Moreover, synergies between the ITER programme and the LFR R&D needs on coolant chemistry and material compatibility are under consideration. In addition, within the frame of the Generation IV International Forum (GIF), international contacts have been established with the developers of the Russian BREST-300 demonstrator and of the US SSTAR concept in order to investigate further synergies and wider cooperation.

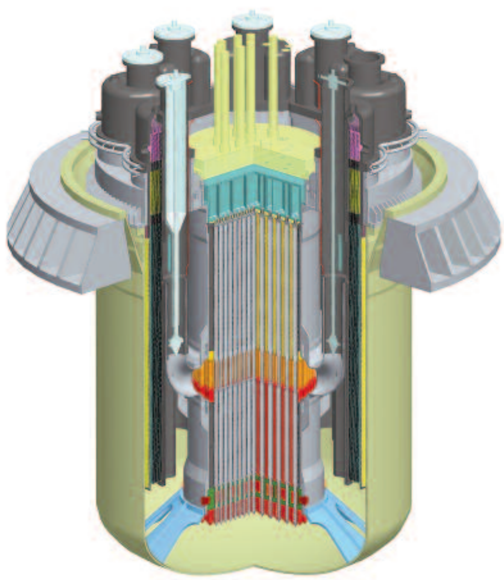

Figure 27: ALFRED reactor (Source: ALFRED Consortium)

Objectives

The Lead Fast Reactor technology is a very promising candidate among the Generation IV Fast Reactors concepts, strictly fulfilling all the main goals as defined by the Generation IV International Forum (GIF). LFR is based on a closed fuel cycle (Sustainability), the inert nature of the coolant provides important design simplification (Economics) and allows for designing decay heat removal systems based on well-known light water technology and passive features (Safety). Moreover, the LFR fuel[19] taken as a reference in the European development programme constitutes a very unattractive route for diversion or theft of weapons-grade materials and provides increased physical protection against acts of terrorism (Non-proliferation and Physical Protection).

19 - MA-bearing MOX (equilibrium concentration of about 17.5 wt.% of Pu and 1 wt.% of MAs) envisaged for homogeneous reprocessing of all actinides for actual closure of the fuel cycle.

In the last decades, the unavailability of qualified materials in a heavy liquid metal environment at relatively high temperatures (above 500 - 550°C) has forced the selection of Lead-Bismuth Eutectic (LBE) as primary coolant for the ADS technology, leading to the MYRRHA project, as the European Technology Pilot Plant (ETPP) among the ESNII initiatives.

ALFRED will represent the first time that a critical heavy liquid metal cooled reactor would provide electricity to the grid

However, the objectives of large scale, sustainable and competitive nuclear energy production are only achievable through a pure lead cooled fast reactor with higher operational temperatures and higher efficiency. Long-term European LFR development will benefit from the safety features already developed for both the MYRRHA and ALFRED projects where the inertness and intrinsic characteristics of the heavy liquid metal coolants have been and will be duly taken into account through specific design provisions. ALFRED represents the first step of the LFR initiative, the European Technology Demonstrator Reactor (ETDR) of the LFR technology, the first plant connected to the grid and fulfilling the Generation IV goals. This would be the first time that a critical heavy liquid metal cooled reactor would provide electricity to the grid.

Besides the different objectives of MYRRHA and ALFRED, it is important to stress the obvious strong synergies characterised by the basic similarities of coolant technologies and further enhanced by the strong collaboration already existing among research centres and industries widely involved in both the ADS and LFR activities as well as in fusion technology. Indeed, the LFR roadmap is based on a number of European experimental facilities dedicated to Lead and Lead-Bismuth technology, and takes advantage of the nuclear data collection and operational experience gained at the Guinevere facility.

As fully described below, the high level of flexibility reached in the design phase of ALFRED, will allow for a short-term deployment strategy as soon as financial instruments are available. In the LFR long term deployment strategy, the ALFRED operation will take full advantage of the gained experience and of the data made available by the above mentioned facilities, including MYRRHA.

State of the Art

Starting from April 2010 the LEADER project carried out an important set of activities with two main goals: the advancement of the conceptual design of the industrial size plant to the present European LFR configuration (ELFR), rated at 600 MWe, and the development of the design of the LFR demonstrator ALFRED, a fundamental step on the LFR roadmap.

The present configuration of ALFRED is illustrated in the following picture:

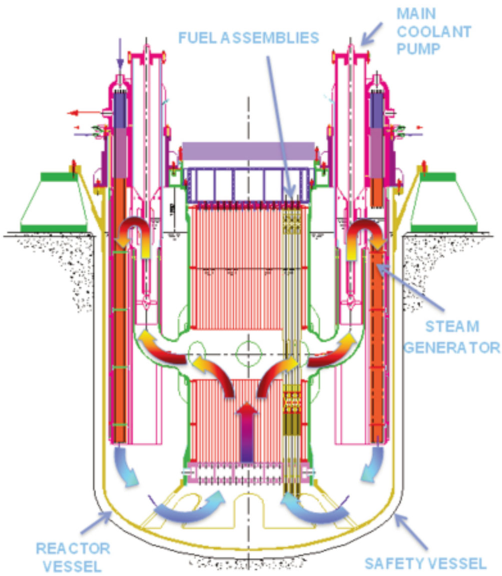

Figure 28: ALFRED Reactor Configuration
(Source: ALFRED Consortium)

Main features of the ALFRED design are:

- pool type configuration characterised by a reactor vessel and the cavity liner safety vessel
- hexagonal wrapped fuel assemblies extended to cover gas to simplify fuel handling (FAs weighted down by tungsten ballast for refueling and kept in position by upper grid springs during operation)
- mechanical pumps
- double-walled straight SG tubes with continuous monitoring of tube leakages
- reference thermal power of 300 MWth

The thermal cycle is completely consistent with the ELFR thermal cycle: primary lead temperature being between 400-480°C, secondary side pressure 180 bar, once-through SGs with water/steam temperature range from 335 to 450°C in superheated conditions, the overall efficiency has been evaluated higher than 42%.

ALFRED will also allow for testing the connection to the electrical grid, with a generated power of about 120 MWe.

The safety of ALFRED is extensively based on the use of the defence in depth criteria, enhanced by the use of passive safety systems (actively actuated through locally stored energy source, always available, and fully passively operated). Safety features of the LFR system have been designed since the beginning of the activities to face challenging conditions and events, thanks to the very forgiving and benign characteristics of the coolant. As an example, there is no need for off-site or emergency AC electrical power supply to manage the design basis accident conditions, the only action needed is the addition of water to maintain the level in the decay heat removal (DHR) pools which are already sized to guarantee at least three days of unassisted fully passive operation and can be easily refilled in the following days.

ALFRED design is conceived in order to maximise flexibility during the operational phase and take full advantage of the experience and data made available in the meanwhile. The operation will take place in two main steps: the first one will be carried out with the available fuel at the time of plant start-up (present choice is MOX), the second one will exploit results of investigations carried out by MYRRHA to implement innovative fuel for the LFR demonstrator reactor. The almost parallel development of the LBE and pure-lead technologies, combined with the two-step approach foreseen for the operation of ALFRED, will allow for a more efficient exploitation of the synergies between ADSs and LFRs, as well as for a broad cooperation and related technologies spin-off.

Such a target is considered technically feasible but obviously needs the allocation of appropriate financial and technical (man-power) resources.

First efforts have been carried out in the past years to provide the necessary basic steps for ALFRED development, namely:

- The activities carried out by the LEADER project related to ALFRED conceptual design.
- The 2011 Romanian proposal to include ALFRED in the country's energy strategy.

- The signature of a Memorandum of Understanding in 2012 by major Italian industry (ANSALDO) and research organisation (ENEA) and the Romanian Research Institute (INR) dedicated to the development of an organisational framework for the ALFRED consortium.

The next step is the formation of an International consortium to advance both ALFRED design and licensing activities.

Besides the coordination activities for ALFRED some technological development to reach a higher level of maturity of the LFR development is still needed. The related activities are summarised below, on the basis of the categorisation developed by the Generation IV LFR System Steering Committee:

- system design and component development
- qualification of materials and development of lead technology
- innovative fuels and fuel cycle

Challenges

System design and component development

The main goals of the LFR system design and component development are:

- approach to control corrosion and erosion of structural materials
- seismic isolators to cope with the large mass of lead
- in-service inspection techniques
- refuelling operations at high temperature (400°C)
- management of Steam Generator Tube Rupture inside the primary system
- prevention of freezing of coolant during all operational states

Research on phenomena of corrosion and erosion by molten lead and their prevention for candidate structural steels for the primary system is essential. For near term deployment, the use of existing, qualified industrial materials for most parts of the reactor equipment is possible, by limiting the core outlet temperature, whereas new materials or specific coatings are being developed for special components, especially claddings. For longer term deployment, approaches beyond the usual “oxygen control strategy” may be explored to extend the operability conditions in terms of temperature range.

The mass of lead is minimised by design features; developments are ongoing through a dedicated FP7 project (SILER) to develop suitable 2-D seismic isolators for the reactor building.

The fuel assemblies are fitted with an extended stem which allows the fuel handling machine to operate in the cover gas under full visibility conditions. This completely eliminates in-vessel fuel transfer equipment.

Steam Generator Tube Rupture is being investigated by experimental tests aimed to demonstrate that such events do not compromise safety, i.e. they can be adequately prevented or mitigated, and will not represent a challenge to the investment protection.

Materials qualification and lead technology development

The strategy consists of two tracks, for short- and medium-term deployment:

- use of existing qualified materials (short term)
- development of the coolant oxygen control for very large pools (short term),
- development of innovative materials and coolant technology (medium term)
- assessment of the possibility for application of material surface coating (medium term)

Due to the large database available, austenitic steels, and especially those of low-carbon grade, are candidates for components operating at relatively low temperatures and low irradiation fluence, e.g. the reactor vessel. Advanced austenitic stainless steel (such as the 15-15 Ti strengthened and its evolutions) in the short and medium term appear to be the most suitable solution for fuel cladding because already proven in SFRs, even if the corrosion resistance in lead still needs to be addressed. The possibility to adopt a coating for this aim is under investigation even if its performance has to be proven by irradiation tests in a lead environment.

For long term deployment, ferritic-martensitic steels appear to be among the best candidate materials for fuel cladding and structures because of their higher resistance against swelling under high fast neutron fluence. Nevertheless several characteristics of ferritic-martensitic steels such as the fatigue softening, DBTT shift under irradiation, type IV weld cracking, creep resistance and thermal ageing still have to be properly addressed.

The resulting R&D needs consist of the qualification of:

- an austenitic steel for the reactor vessel
- lead corrosion resistant material for the steam generators tubing
- protective coating for fuel cladding and fuel element structural parts
- special materials or coatings for the impeller of the mechanical pumps

The use of lead coolant implies also:

- development and validation of a technique for lead purification (prior to use and online during operation to recover activated corrosion products or e.g. volatile Hg)
- development and calibration of instrumentation operating in lead and under irradiation (fuel cladding detection instrumentation, coolant chemistry control, thermal-hydraulics monitoring, ultrasonic instrumentation under liquid metal…)
- development of techniques and instrumentation for in-service inspection
- development of a waste management strategy for used lead

Thus, the strategy for material qualification and lead technology development consists of a two-step approach based on the need to achieve demonstration in the short term and optimise the system for long-term industrial deployment. Consequently, as mentioned for MYRRHA, ALFRED will use already available technology (e.g. 15-15Ti for the cladding without or with surface coating) while the ELFR can fully exploit the advantages of innovative materials (innovative austenitic steels, T91, or ODS - which however still needs irradiation qualification) and of possible innovative breakthroughs regarding lead coolant chemistry and its purification. Special attention is presently dedicated, for this purpose, to the austenitic and ferritic-martensitic steels containing Al and/or Si as alloying elements, due to their high resistance to lead corrosion, even though special attention will be given to irradiation embrittlement.

Innovative fuels and fuel cycle

Fuel development from the demonstration phase to industrial deployment:

- ready-to-use technical solutions for demonstration in the near term
- mid-term goal to confirm the use of MA bearing fuels
- development of innovative fuel solutions for industrial deployment

In the near term, an essential step of the LFR development is the availability of ready-to-use technical solutions, so that fuel can be provided on time tested, and qualified, with the parallel development of suitable performance codes...

The LFR R&D programme presents strong synergies with the SFR fuel development activities, as recognised by both communities

In the mid term, it is necessary to confirm the possibility of using advanced MA (Minor Actinide)-bearing fuels and the possibility of achieving high fuel burn-up.

In the long term, it is important to confirm the potential for industrial deployment of advanced MA-bearing fuels and the possibility of using innovative fuels having higher conductivity and lower swelling that can withstand high temperatures thus increasing fuel safety margin.

The R&D programme presents strong synergies with the SFR fuel development activities, as recognised by both communities. Due to the large number of similarities and common activities, the identification of a common line of development for both systems is of mutual interest and, consequently, strongly suggested. Qualification of the cladding material, being a very long and expensive task, may also take advantage of the research programmes carried out for both SFR and LBE systems resulting in very important savings in terms of overall cost and efficiency. Other cross-cutting research activities have been identified in the fields of core safety, fuel safety, seismic studies as well as instrumentation, inspection and repair techniques.

The ALLEGRO Project

ALLEGRO is the Gas cooled Fast Reactor (GFR) demonstrator as identified in the roadmap for the development of the Gas Fast Reactor technology.

Objectives

The GFR cooled by helium is proposed as a longer term alternative to sodium cooled fast reactors (SFR). As well as offering the advantages of improved inspection, simplified coolant handling and low void reactivity, the GFR offers the unique advantage of fulfilling two missions:

1) closure of the nuclear fuel cycle and simultaneously providing a sustainable nuclear energy source as with other ESNII-concepts
2) delivery of high temperature heat at ~800°C (process heat, production of hydrogen, synthetic fuels…)

The helium cooled Fast Reactor is an innovative nuclear system having attractive features: helium is transparent to neutrons and is not chemically reactive. Its viability is essentially based on:

- the development of a refractory and dense fuel
- robust management of accidental transients, especially after the Fukushima accident

> The GFR cooled by helium is proposed as a longer term alternative to sodium cooled fast reactors

For GFR to become an industrial reality, an intermediate objective is the design and construction of a small demonstration reactor. This reactor has been named ALLEGRO and its role, apart from being the world's first gas cooled fast reactor, consists of the following:

- pilot scale demonstration of GFR-specific safety systems taking benefit from simpler in-service inspection and repair and coolant management
- final qualification of the innovative high-temperature (ceramic) fuel at the full core level required for GFR
- testing of the GFR-related technologies such as e.g. refuelling, spent fuel reprocessing and refabricating, helium purification & regeneration, high-temperature materials, GFR-related components
- potential test capacity of high temperature components or heat processes

State of the art

A carefully planned and extensive R&D of GFRs started after 2001 in France at CEA and continued on the GFR 2400 MWt and ALLEGRO 75 MWt concepts until 2009, when the GFR programme was reduced. International collaboration of CEA with other European institutions took place (or is still underway) mainly within the EURATOM Framework Programmes (FP6 GCFR STREP, FP7 GoFastR).

In 2010, three research institutes from the Czech Republic, Hungary and Slovakia, stepped into the ALLEGRO development, with the aim of creating an ALLEGRO Consortium and hosting the demonstrator in one of these countries. A Memorandum of Understanding was signed on 20 May, 2010 between ÚJV Řež, a.s. (Czech Republic), MTA-EK Budapest (Hungary) and VUJE, a.s. (Slovakia). The National Centre of Nuclear Research (NCBJ) Warsaw (Poland) signed the Memorandum of Understanding in 2012 as associated member. The CEA contributes to the preliminary phase of the project. Consecutively the formation of the international Consortium is underway.

The Consortium members agree to use their own financial resources in combination with the expected governmental support in their countries and international support from the EU Framework. The Consortium assumes the establishment of a GFR Research Centre of Excellence. It is worth mentioning that the Czech and Slovak republic (former Czechoslovakia) built and operated a gas (CO_2) cooled heavy-water moderated nuclear reactor KS-150 in the period 1972-1977.

The demonstration of the GFR technology assumes that the basic features of the 2400 MWt GFR reactor can be tested in the 75 MWt ALLEGRO. Therefore, most of the main parameters of both reactors are similar to each other (power density, etc.).

The current CEA Concept of ALLEGRO is characterised by a metallic reactor pressure vessel (RPV), upward core cooling, control rod mechanism through the RPV bottom entry, two primary loops (realised as a coaxial cross-duct) and two circuits (the primary helium and the secondary water). Three decay heat removal (DHR) loops containing water-cooled heat exchangers located well above the core represent another important feature of the Concept. The primary circuit of the CEA Concept is shown in the following figure 29.

As the production of electricity is not the primary goal, the CEA Concept has no power conversion system.

The primary circuit is filled with helium pressurised to 7 MPa. The whole primary circuit is integrated, including the DHR loops, in a

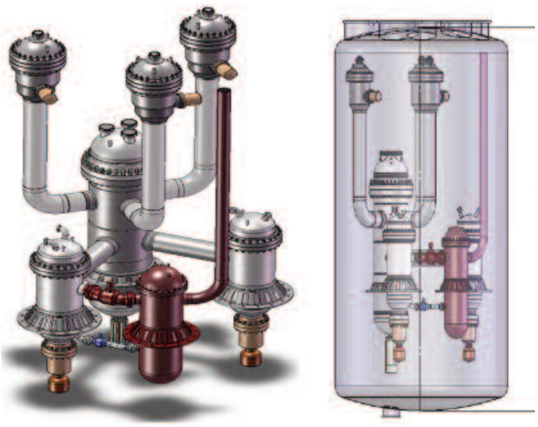

Figure 29: CEA Concept of the ALLEGRO reactor (Source: ALLEGRO Consortium)

cylindrical close containment call the guard vessel, which is filled with atmospheric nitrogen. The water in the secondary circuit (not shown in the Figure) is pressurised to 6.5 MPa. The red gas/gas heat exchanger in the Figure 29 is planned for extraction of process heat.

Since ALLEGRO will be a demonstrator of the GFR2400 concept, the development, testing, and qualification of the advanced fuel (applicable in the GFR2400) is of primary importance.

Two successive core configurations are, therefore, expected. The starting core will be based on MOX fuel containing ~25% Pu in stainless steel cladding. This fuel will be derived from the SFR programme and will serve as a driving core (Tinlet/Toutlet He 260/530 °C) for six experimental fuel assemblies containing the advanced ceramic fuel (pin-type mixed carbide fuel in SiCf/SiC claddings resistant in accident conditions up to 1600 °C for few hours), Figure 30 Flow reduction in these assemblies will enable to reach ~850 °C at the outlet from these assemblies. The pressure drop in the core is designed to be below 0.15 MPa to ease the gas circulation. The final core of ALLEGRO will consist solely of the ceramic fuel and will enable the operation of ALLEGRO at the high target temperature (Tinlet/Toutlet He 400/850 °C).

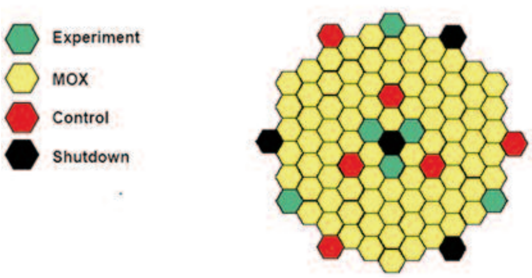

Figure 30: The 75 MWt ALLEGRO MOX core (Source: ALLEGRO Consortium)

To maximise the similarity with the GFR2400 and to increase safety, the Consortium proposed an intermediate gas circuit filled with He+N2 mixture to insert into the scheme of the CEA Concept. The second reason was to minimise the risk of a massive water leakage into the primary circuit filled with hot helium (corrosion, criticality). A gas expansion turbine was proposed for this circuit by CEA as a safety feature just to produce power for the blowers in case of a blackout. The third circuit, conducting the heat to the cooling tower contains alternatively a gas/water heat exchanger or a steam generator for a power conversion system represented by a small steam turbine, operated in a Rankine cycle. The use of both the steam turbine as well as the gas expansion turbine for ALLEGRO is still under discussion.

To support the theoretical R&D, several experimental facilities were designed at CEA. Some of them were constructed and started to generate data already before 2009. An integral high-temperature helium loop for testing of the DHR system, components and code validation, however, has not yet been built. The existing experimental facilities for GFR-related research were summarised within the FP7 project ADRIANA in 2011.

Challenges

The challenges are mainly related to the demonstration of safety, the fuel technology able to withstand high temperatures, the material issues, and the helium-related technology.

Optimisation of the design for ensuring cooling of the core in accident conditions

During a loss of external power for blowers in the gas circuits (especially during loss-of-primary-coolant accidents associated with significant depressurisation), forced convection is required for successful removal of the decay heat. CEA proposed to take advantage of the above mentioned gas turbine in the intermediate He+N2 circuit driven by the decay heat transferred from the primary circuit through the gas/gas heat exchanger into the intermediate circuit. This turbine is expected to drive the blowers in both gas circuits in accident conditions. Analyses of this option as well as the assessment of technological feasibility have to be performed.

Qualification of the GFR-related DHR approach

The behaviour of the GFR-specific DHR system, i.e. additional water-cooled heat exchangers located in loops well above the core, was simulated numerically but not yet experimentally. An integral loop for a complex test of the DHR approach is planned at CV Řež (Czech Republic) to test the following phenomena:

- qualification of the GFR-related DHR approach
- validation of system codes (e.g. CATHARE or those under development in HTR projects)
- capability to switch the core cooling from the main loops to the DHR loops and their capability to operate in natural circulation in expected conditions
- capability to avoid core by-pass in LOCA conditions especially under interaction of several main & DHR loops

Development of the carbide (U,Pu)C fuel in SiCf/SiC cladding for the second core

CEA evaluated this type of fuel as a promising option for the high-temperature GFR core and achieved a significant progress in its development. Priority was given to pin-type fuel. The plate-fuel, originally considered as a very promising option, was abandoned. The following R&D is expected:

- further optimisation of the SiCf/SiC design, properties, performance, cost, and technology (component production route, plug technology, hermetic sealing using a suitable liner material, irradiation-enhanced creep, oxidation by impurities in the helium coolant, embrittlement by irradiation at low temperatures)
- out-of pile and in-pile testing of the mixed carbide fuel & SiCf/SiC segments (optimisation & assessment of porosity, thermal conductivity, etc.)
- minimisation of the fuel-cladding mechanical & chemical interaction
- assessment of the SiCf/SiC abrasion/erosion in flowing helium containing impurities.
- development & validation of the models for swelling and fission gas release from the mixed carbide fuel in operational conditions
- implementation of mechanical & physical properties into fuel behaviour codes and their validation for the planned operating domain with respect to temperatures, burn-up, etc.

Mitigation of severe accidents (SA)

Efficient design features for mitigation of SA are expected to be proposed and implemented into the ALLEGRO concept. Thorough analyses using suitable severe accident codes should prove that WENRA requirements would be fulfilled for both the MOX and the ceramic cores. This concerns at present e.g. the assessment of so called unprotected accidents.

The behaviour of the molten fuel and its coolability, have to be studied.

Development & qualification of the wire-wrapped MOX fuel for the first core

The MOX fuel from the SFR programme needs to be fabricated and qualified for its use in helium coolant with prototypical GFR parameters. This includes also the optimisation of both the cladding material and the hexagonal wrapper tube as well as the extension of the fuel performance codes to the ALLEGRO fuel (validation of the fission gas release model, assessment of temperature non-uniformities in the MOX bundle, heat exchanges, pressure drop, provision of various physical properties e.g. heat transfer into helium, material properties of both MOX and cladding).

Development of thermal barriers and insulation materials

Thermal barriers are needed in the GFR design to protect structural metallic materials from excessive temperature load:

- thermal barrier (panels) protecting the inner surface of the reactor pressure vessel (goal to withstand short term 1250°C in accident conditions)
- thermal shield of the experimental (U,Pu)C assemblies in MOX core (wrapper tube)
- insulation of the hot duct in the coaxial piping

Other GFR-related technologies

- helium purification system for gas circuits to limit activated impurities and corrosion
- tritium management
- Regeneration of filters
- helium recovery from the nitrogen guard vessel gas (helium economy)

- sealing technology (goal to reduce the leakage rate to 10% of He inventory per year)
- wear resistance of materials, especially ceramic thermal shields and insulation
- management of helium leaks to containment

Qualification of GFR-specific components including material issues, especially

- gas/gas heat exchangers - design and heat transfer coefficients
- active and passive valves (reliability and ageing)
- fuel handling system
- control rods - absorber & cladding material, drive mechanism
- reflector around the core – qualification of the material, mechanical & thermal properties
- instrumentation (e.g. optical measurements of temperature)

The above list is indicative only; there are other issues that need further attention such as e.g. core physics (voiding reactivity effect, qualification of neutron leakage through axial gas channels, neutronics/thermal hydraulic coupling).

Some phenomena are still not modelled today:

- sub-assembly, core and collector thermal-hydraulics
- core mechanics (equilibrium and seismic response)
- transport of contamination

NC2I - Nuclear Cogeneration Industrial Initiative

Nuclear cogeneration relates to the co-production of heat and electricity using a nuclear reactor. Fossil fuels are today by far the main source of heat for European industry, transport and households. The production of heat with nuclear technology is a major innovation that can open a new and significant market potential for nuclear systems, whilst providing a significant contribution to European strategic energy policy in terms of curbing CO_2 emissions and increasing security of energy supply.

Introduction

Rationale

Although European industry has achieved impressive emission reductions in recent years, fossil fuel combustion to provide heat to the processes of European energy intensive industries corresponds to an annual emission of 720 $MtCO_2$. This represents around 20% of Europe's CO_2 emissions. This is more than the 470 $MtCO_2$ emitted every year to generate the electricity consumed by these industries.

Energy is vital to EU industries and accounts for a significant share of the production costs of EU energy intensive industries, with for instance 8% for chemicals production or 22% for pulp & papermaking. European industry represents a large contribution to the European economy and generates essential products for our everyday lives. To maintain and strengthen this position in Europe, a low-carbon, competitive energy technology is needed. Nuclear cogeneration can be the feasible innovation to meet that requirement.

Additionally, short-term opportunities for households such as district heating, and desalination to solve fresh water shortages, and long-term opportunities for reducing fossil resource usage in transport by generating synthetic transport fuels via nuclear powered hydrogen production, add a significant market and carbon emission reduction potential.

The EUROPAIRS project has analysed in depth the European industrial heat market. The vast market potential for nuclear cogeneration was highlighted, most recently in Europe (EUROPAIRS project) and in the USA (MPR Associates), cf. Table 1.

Region	Heat market estimation	Of which produced by cogeneration
Europe	3 000 TWh/y (EUROPAIRS)	800 TWh/y [22]
USA	3 600 TWh/y (MPR Associates)	1100 TWh/y [23]
Rest of the world	N/A: 10 000 TWh/y	N/A

Table 1: Heat market by region [22 23]

Various nuclear reactor technologies could each meet a part of the market demand as shown in Table 2.

State of the Art

Nuclear cogeneration is already a reality. In 2006, 1700 reactor years of experience has been accumulated worldwide, mainly for waste heat valorisation from water reactors[20 21]. In Europe, more than 1000 GWh of low-temperature nuclear heat was produced in 2006 in Bulgaria, Czech Republic, Hungary, Romania, Slovakia and Switzerland.

Water reactors have an extensive operational experience, including in low-temperature cogeneration. Low temperature cogeneration from a fast neutron reactor was proven in one Kazakh plant (BN-350) for desalination[20]. Significant development is however needed

20 - Advanced Applications of Water-Cooled Nuclear Power Plants, IAEA-TECDOC-1584, July 2007.

21 - Advances in Nuclear Power Process Heat Applications, IAEA-TECDOC-1682, May 2012.

22 - Bredimas, Market study: Energy usage in European heat intensive industries, Executive summary, LGI Consulting report ET_1103, May 2011.

23 - Survey of HTGR Process Energy Applications, MPR report MPR-3181, May 2008.

before cogeneration application can be considered for medium level temperature applications. As time progresses and fast reactors are further developed, the obstacles for fast reactor cogeneration applications will diminish.

High temperature reactors provide significant perspectives for medium and high temperature cogeneration applications. The HTR technology builds on the developments in Germany in the 1980s, as well as previous research in UK and USA, re-established and revived in several national and European Framework Programme projects from the year 2000 onwards, of which the ARCHER project (2011-2014) is currently ongoing.

The coupling with end-users of HTR for high temperature cogeneration has still to be developed[21]. The EUROPAIRS project has established direct contacts between the conventional process industry and the nuclear community and has developed key performance indicators. It has also identified operational envelopes for the coupling and assessed the general licensing aspects on dedicated case studies (hydrogen, refining). Additionally, the establishment of an HTR demonstrator coupled to industry has been regarded as essential by the industry in EUROPAIRS, to enable market breakthrough by risk reduction and the more reasonable deployment horizon of demonstrator follow-ups.

The ARCHER project provides technology R&D in support of demonstration, in which both fundamental research and direct applications are combined. This basis should be further extended in strategic directions, to minimise demonstrator risks, by securing the licensing framework especially.

National projects are supporting the European R&D. For instance, Polish project HTRPL was launched in September 2012 to strengthen the national scientific and technical capacities for the HTR programme in Poland. It gathers universities, research institutes (covering nuclear, fertilizers and coal processing R&D fields), an engineering company, a power plant operator and an energy intensive company. The German project SYNKOPE was launched in

24 - Heat Loads and Polygeneration Applications - Chemical, food, paper and refinery sectors, D-Ploy Workpackage 2 deliverable, August 2008.

Reactor type	Max. potential steam temperature	Relevant applications	Minimum temperature range needed	Comment	Total EU heat market
Water reactors	250°C	District heating	70-130°C	Seasonal, nuclear already used	700 TWh/y
		Desalination	100-130°C	High growth expected, nuclear already used	
Fast breeders	450°C	Pulp & paper	100-250°C	High load volatility, nuclear already used	
High temperature reactors + GFR	550-750°C	Oil refining	500-550°C		1000 TWh/y
		Chemicals	500-550°C		
		Fertilizers	500-600°C	Largest H_2 application	
Very high temperature reactors	1000°C	Hydrogen	750-850°C	Sector expected to grow, in particular with new applications in transport and energy	1300 TWh/y
		Other applications	1000+°C	Other sectors could be envisaged for nuclear pre-heating (e.g. lime, aluminium, iron & steel, high-temperature O_2 production, glass)	

Table 2: Heat market per type of nuclear reactor technologies [21 22 24]

August 2012 and elaborates nuclear-assisted coal-to-liquid process scenarios. Additionally, the German project STAUB-II launched in August 2012 is to work on HTR safety, including experiments in a newly built helium test facility.

Main Design Options for HTR

Two HTR design concepts use the same high safety standard TRISO fuel particles either embedded in graphite spheres for the pebble bed core or in compacts inserted in graphite blocks in the block-type core. The present major representatives of the design are the test reactors HTR10 (pebble bed) in China and the HTTR (block-type) in Japan. The power rating for the modular HTR has been evaluated between 200 and 600 MWth with an average steam temperature of about 550°C, in line with the current conventional steam cycle technology.

The HTR design allows high flexibility in terms of power rating and temperature. In addition, its inherent safety characteristics including limitation of fuel temperature in case of accidents and the threefold containment of radioactivity in the TRISO fuel particles complemented by the inert helium coolant demonstrate the high safety standard of HTR[25].

The design options can be classified into:

1. **Short term**: indirect cycle, steam production 550-600 °C (current coal fired power plant conditions as reference), power split depending on demand
2. **Mid term**: follow materials development towards higher temperature applications in fossil-fired plants, possibly switch to heat carrier other than steam
3. **Long term**: 950°C or beyond (primary side) requires change and thus development of structural materials for applications such as thermo-chemical H2 production and other high temperature processes

Taking into account the favourable safety characteristics of HTR, the coupling of an HTR to a conventional process has to take into account the different licensing procedures and the avoidance of negative impact of the nuclear energy source and the industrial process side, and vice versa.

An example as proposed by AREVA for the US demonstrator project NGNP is shown in figure 31:

25- Advances in High Temperature Gas Cooled Reactor Fuel Technology, IAEA-TECDOC-1674, June 2012.

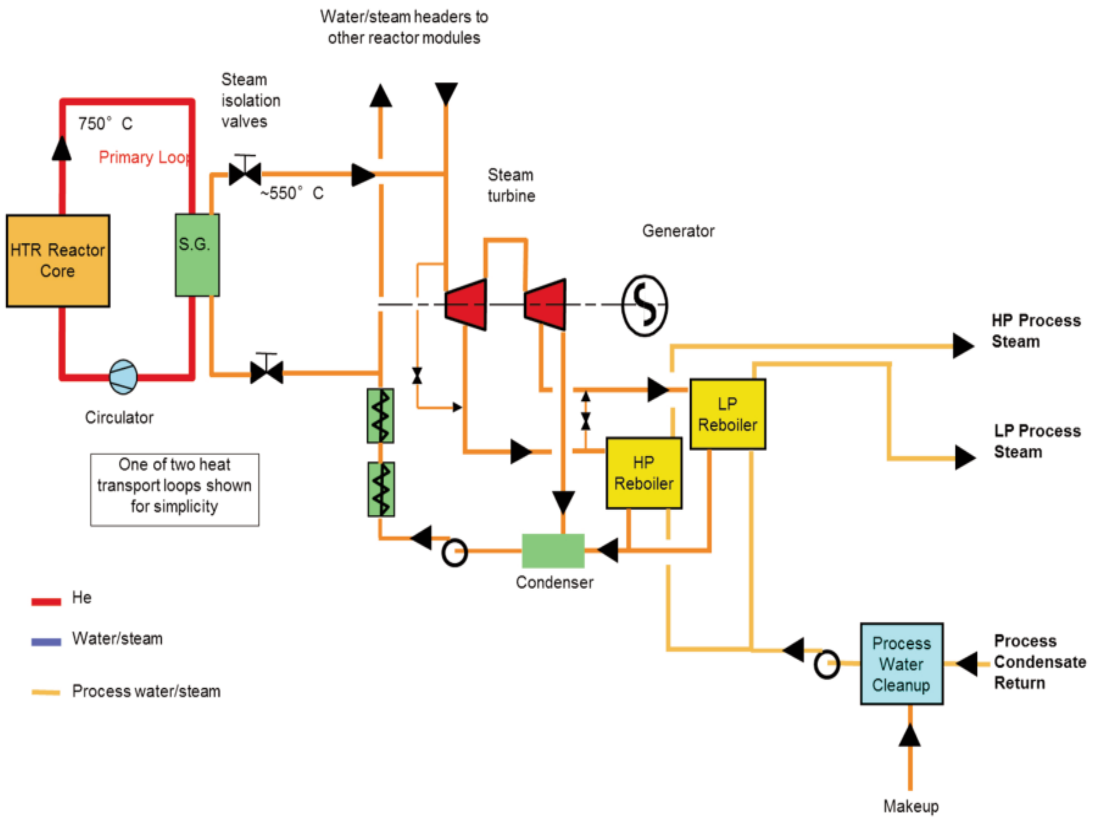

Figure 31: Commercial Process Heat Cogeneration Facility Basic Configuration (AREVA proposal for NGNP)

Challenges: Research, Development and Innovation

In this chapter the R&D efforts are summarised that would stimulate and facilitate nuclear cogeneration implementation. First, generic items are named that are of importance for all nuclear systems in cogeneration mode. Secondly, the most relevant R&D towards demonstration of nuclear cogeneration using high temperature reactor (HTR) technology, with its favourable characteristics for cogeneration applications, is listed. Thirdly, R&D is outlined that would further broaden the potential of HTR technology. This is at this stage considered more long term, but should be initiated in the short term to enable the strengthening of specific long-term deployment scenarios and applications.

Generic R&D

The following nuclear cogeneration R&D subjects are generic, as they are relevant for all nuclear systems operating in cogeneration mode. However, their importance and relevance depend strongly on the nuclear system envisaged, and the process it is connected to. These mainly relate to safety and licensing but also to the exploitation of existing LWR for purposes of growing importance (e.g. seawater desalination):

- **tritium transport reduction to secondary and tertiary systems:**
  - tritium permeation barriers
  - coolant chemistry and purification technology
- **impact of process transients on cogeneration supply unit, and vice versa:**
  - coupled system code development and validation
- **coupling technology including energy buffering:**
  - coupling component development and qualification
  - coupling material selection and performance qualification
- **adaptation of existing LWR and future SMR to meet strongly growing demand for district cooling and seawater desalination in arid countries**

Non-cogeneration specific R&D for large water-cooled reactors, water-cooled small and medium-sized reactors (SMR) and for fast neutron reactors is referred to in the SRIA chapters associated with these technologies.

High Temperature Reactor related R&D towards demonstration

The HTR features high efficiency potential due to elevated primary coolant temperatures and a very high level of inherent safety. These features make the system particularly appropriate for the process industry (e.g. chemistry) or other high temperature applications, for which a very efficient and CO_2-free energy source is a favourable alternative[2].

The HTR features high efficiency potential due to elevated primary coolant temperatures and a very high level of inherent safety

Demonstration is required for HTR technology market breakthrough. The following topics are essential for demonstration to succeed within reasonable timescales and with minimised risks, and the R&D efforts associated are listed for each topic specifically:

Safety demonstration

- **accident modelling (air and/or water ingress)**
  - thermal-hydraulic and fuel performance code development and experimental validation for accidents and transients
  - integral safety testing: water and air ingress experiments such as NACOK and safety tests under irradiation
- **HTR earthquake response (graphite core and component vibration analysis), identification of suitable countermeasures, e.g. design innovation to minimise loads on supports, flanges and connectors**
- **response of nuclear heat source to transients from heat sink and vice versa (coupling impact)**

Licensing support

- **renewal and adjustment of the past German HTR licensing process to current standards**
- **support the development of HTR oriented European design codes and standards**
- **fuel and graphite back end: compatibility of HTR waste with final repository conditions**
- **source term determination (fission product transport)**

- develop probabilistic licensing approach with regards to HTR fuel and core components
- develop a system integration approach for the HTR system, and of the coupled system, and apply this to actual application cases
- develop, prepare and perform the material and fuel characterisation and qualification process.

Technology innovation

To maintain and strengthen the HTR knowledge base, support demonstration, possibly as international collaboration:

- helium technology (purification, primary system sealing, pumps, circulator)
- material behaviour in a helium environment (tribology tests, helium loop, helium flow erosion/corrosion)
- hot duct design
- (digital) high temperature Instrumentation and Control for online system performance and behaviour monitoring and analysis
- fuel and graphite hardening against air and water ingress
- heat exchanger and system design to minimise air and water ingress
- fundamental understanding of HTR fuel performance and behaviour by appropriate (separate effect) irradiation tests and modelling efforts, to determine:
  - material properties and behaviour under irradiation
  - fission product transport and diffusion
  - manufacturing impact on fuel performance
- adopt the fundamental knowledge regarding HTR fuel behaviour, develop and validate a mechanistic fuel performance code, including transient behaviour and severe accidents

R&D towards deployment

Beyond demonstration, the potential of nuclear heat sources can be further broadened by appropriate R&D. Of particular relevance is the research in the following areas:

- For HTR: increased primary coolant temperature (950°C) for enhanced efficiency and broader application perspectives:
  - development and qualification of suitable structural materials
  - fuel behaviour and performance, and potential improvement to maintain HTR safety margins
  - coordination with non-nuclear projects on hydrogen production technologies
- For LWR (incl. SMR): maximise the short-term usefulness of low temperature cogeneration applications such as heating/cooling and seawater desalination:
  - Design, test and qualify passive heat removal systems and minimise external cooling requirements.
  - Develop components and scale up technologies to use LWR or SMR for district cooling and drinking water supply in arid areas.
- Alternative fuel cycles, including thorium, to stretch fuel resources, to minimise waste and to optimise cycle length: cf. related SRA section on Fuel Cycles.

R&D infrastructure: the bridge to deployment

Besides the essential competences and experience of experts involved in HTR development (a critical asset for the whole nuclear sector), major investments are needed in modern experimental infrastructure and facilities to enable the above R&D to be performed adequately. The following facilities are essential:

- New irradiation facilities for the investigation, characterisation, development and, ultimately, validation and qualification of HTR fuels and materials. Accident testing requires development of novel and specific irradiation test facilities and equipment. Additionally an in-pile helium loop is needed to assess material behaviour under representative primary coolant flow and irradiation conditions.
- Out-of-pile testing facilities such as accident tests (NACOK, KORA), helium loops (HELOKA, HEFUS).
- Modern hot cells with heating tests (KÜFA) and state-of-the-art PIE possibilities to enable the generation of the appropriate data for code development and validation, and to increase the fundamental understanding of material and fuel behaviour.
- Fuel manufacturing laboratory, also able to handle transuranics.

These facilities serve two purposes: on the one hand they are essential for appropriate R&D to be performed, on the other, they form the basis for the design, licensing and operation of a demonstration project.

The steps towards demonstrator construction go partly beyond R&D and are therefore considered beyond the scope of the SNETP Strategic Research Agenda, but they are closely tied, and therefore shortly mentioned here. As the EUROPAIRS roadmap has shown, the following critical items have been identified for a demonstrator to be established:

- **HTR fuel pilot plant, or qualified fuel delivery from external supplier**
- **manufacturers and suppliers, and the appropriate qualification and quality assurance fitting HTR technology and licensing requirements for:**
  - reactor pressure vessel (most critical)
  - other components, such as heat exchanger, graphite, circulator, hot duct, core structure, steam generator, civil works

R&D to bring innovation to the market

Nuclear cogeneration is a technologically innovation with a potentially major positive impact on European energy policy, supporting curbing, increasing security of energy supply and strengthening the position of European industry, which represents a key economic asset in Europe.

The R&D presented in this chapter is directed towards bringing this innovation to the market via the shortest routes.

Low temperature cogeneration requires minor R&D, which revolves around licensing and coupling technology and safety.

Medium to high temperature cogeneration focuses on high temperature reactor technology given its favourable characteristics and very large deployment potential. Demonstration is worthwhile to achieve this goal, and the R&D can therefore be summarised by the topics: licensing support, safety demonstration and (evolutionary) technology advancement.

Very important is the support which R&D can provide by maintaining and strengthening the nuclear fission knowledge base, by developing and establishing the R&D infrastructure that can bridge the transition from technology innovation to implementation.

Very high temperature reactor technology is a major innovation with a longer term development required, together with alternative fuel cycles.

Demonstration is a prerogative, but the development would span such a long time that, assuming that the perspective can be substantiated, these developments should be initiated in the short term.

Cross-cutting R&D topics

In the present SRIA, the R&D topics have been organised according to the related reactor technologies, however some topics have an intrinsic cross-cutting nature. This is the case for fuel cycle technologies bridging different reactor generations, in particular fuel reprocessing. Education & Training and Knowledge Management are also topics affecting all nuclear technologies. Knowledge management is essential as it allows the storage and dissemination of the results of research

Technical topics addressed as cross-cutting in the SRA 2009, like materials, pre-normative research, simulation tools and infrastructures have been integrated into the three main technological chapters.

■ Fuel reprocessing

As has been described in the Fuel Cycle chapter, many opportunities to improve the optimal utilisation of natural resources and nuclear waste minimisation are opened up by the reprocessing of the nuclear fuel after its use in nuclear reactors. Fuel reprocessing allows the separation of materials that can be reused in thermal or fast nuclear reactors, either to produce additional energy or to minimise the final waste to be sent to the geological repository. Indeed, the reprocessing of used nuclear fuel is a critical component of all the strategies for long-term sustainability of nuclear energy.

The reprocessing of used nuclear fuel is a critical component of all the strategies for long-term sustainability of nuclear energy

Reprocessing of the fuel used in the present LWRs is common industrial practice in France, and similar technologies are also available in the UK. The plutonium and uranium recovered are partially recycled in the same LWRs in the form of MOX and the rest is saved for use in future FNR.

The challenges for R&D in fuel reprocessing, include the industrialisation of laboratory technologies for separation of minor actinides from the high level waste from the reprocessing of the fuel used in present reactors; the development of reprocessing of advanced fuels foreseen for future reactors (FNR, ADS, advanced thermal reactors and HTR); technologies able to perform joint extraction of several actinides; and the minimisation of secondary wastes in all these strategies. These developments should be performed coherently with the technologies for advanced fuel fabrication and characterisation.

National and international initiatives with strong participation of SNETP members, including the EURATOM ACSEPT project, have been working on reprocessing and conditioning of LWR and advanced fuels for MA separation. In particular that project has provided progress on the definition of extraction molecules and the concept for hydro-metallurgical process with MA separation. A new project has already been approved by EURATOM to study the stability of these molecules under realistic radiological and chemical conditions and to assess the industrialisation of the process. The involved organisations continue undertaking more fundamental R&D on pyro-metallurgical processes. In parallel, some national studies and others involving several EU countries continue developing the processes for dissolution of MA-bearing MOX and MA targets using a different basic matrix.

In the short term, the required R&D for nuclear waste reprocessing can be performed in several existing basic science and validation facilities, but in the medium term demonstration plants for the reactors, fuel fabrication and advanced reprocessing technologies will be needed, both at national and European (joint) levels.

The priorities for short-term R&D in fuel reprocessing are:

- advanced reprocessing of LWR and advanced fuels for MA separation, using either hydro- or pyro-metallurgical processes
- dissolution of MA-bearing MOX and carbide fuels for FNRs and of MA bearing targets (U-free or UO_2 matrix)
- conversion processes after the separation steps and prior to the re-fabrication of fuels/targets
- processes for HTR fuel recycling and waste reduction, integration of fuel cycle with LWR and FNR
- synthesis of new fuels and their performance assessment, oriented to their reprocessing
- irradiation behaviour of MA-bearing MOX and carbide fuels, and MA bearing targets and dedicated PIE programmes

In the medium term the R&D will need demonstration facilities. The decision to develop or not demonstration facilities for fuel fabrication and reprocessing should be taken in about 2017 depending on the results of the previous steps and of the availability of equivalent facilities in Europe.

In the long term, the R&D should focus on the industrial implementation of partitioning and transmutation

Within its programme to operate a sodium cooled fast reactor prototype by 2020, France is considering the construction of a facility devoted to the manufacturing of the core fuel for the ASTRID FR prototype. This facility, called AFC, could also provide fuel fabrication services for the testing and demonstration of alternative reactor technologies at the European level.

For recycling ASTRID fuel, several options are under consideration: either a dedicated pilot-scale facility (ATC), or adaptation using complementary steps in a LWR fuel reprocessing plant.

Meanwhile an investigation is under way in order to evaluate the possibilities of increasing the capacities of existing facilities such as ATALANTE (CEA/Marcoule). Demonstrative transmutation experiments, at sub-assembly scale, would call for new facilities, able to manufacture MA-bearing targets or MA-bearing fuels, at kg scale. A reflexion should be encouraged within the EU for the design of such facility.

The objectives of these facilities should be to prepare the next generation reprocessing plants, which will be needed for the fast reactor cycle. They should address issues such as multi-recycling of plutonium and minor actinide separation.

In the long term, the R&D should focus on the industrial implementation of partitioning and transmutation: The implementation of this phase will depend on the results of the previous phases and will be mainly carried out under the control of the nuclear industry.

Non MOX fuel still needs some reprocessing development.

Education, training and knowledge management

i - Scope and Objectives

Qualified human resources for both the nuclear industry and nuclear regulation are a crucial prerequisite for the safe and economic use of nuclear fission energy. Due to the stagnation of new build NPPs observed in many areas of the world after the Chernobyl accident, the availability of those resources became an issue, and the low attractiveness of nuclear professions, particularly for the best brains, developed into a key concern in many countries. The impact of the Fukushima accident could aggravate the situation.

However training programmes are needed to maintain European expertise in safety and are already of paramount importance, especially knowing that increased safety precautions are being included in the daily operation of our nuclear power plants and for new build projects.

Nuclear education and training (E&T) is therefore, since many years, a high priority topic on the agenda of many initiatives addressing the use of nuclear fission energy for a sustainable energy supply. Of particular relevance in that respect is the relationship between nuclear education and training and nuclear research: first, the quality of nuclear research directly depends on the interest and engagement of highly qualified scientists and engineers in those activities. Second, research plays a crucial role for the qualification of young scientists and engineers by providing know-why and other important competences required to solve relevant technological and safety issues and to ensure the capability for leadership in organisations involved with nuclear energy issues.

This section of the SRA discusses the recent achievements in nuclear education and training with special regard to the links with nuclear fission research. It further addresses the challenges of the present situation which is, after the Fukushima accident, marked by increased uncertainties about the future development of the nuclear sector in some countries. Last but not least the section will indicate priorities for further actions.

ii - Progress to date

There have been many important actions in the EU member states and at the European level to improve the situation of nuclear E&T. Since the mid-1990s initiatives were started, for instance, to maintain infrastructure and personnel for nuclear education at universities as well as to strengthen the role of nuclear fission research as a tool for nuclear education and training. Particular initiatives worth mentioning are:

- Efforts of the European Commission, both through Euratom supported projects and direct actions carried out by the JRC, and some European member states to maintain research in nuclear fission technology and nuclear fission safety with close links between research and educational organisations.
- Support by the industry, regulators, and research centres for universities in maintaining academic education in nuclear technology, when the number of students was not sufficient to warrant financial support from government and/or education authorities.
- Building of national and international networks with the objective of strengthening the cooperation between different universities, promoting cooperation between universities, research centres and other nuclear stakeholders, facilitating the exchange of information, collaboration and the sharing of best practices in nuclear education and training, and making studies in nuclear energy more attractive for students.
- Establishment of European and national data bases providing information about E&T opportunities and employment offers.
- Creation of new national and international training programmes - some of them with substantial engagement of employers - devoted to an effective preparation of engineers and scientists for new jobs and/or new positions in the nuclear industry.
- Creation of bodies investigating the structure of and the future demand for nuclear professionals on a regular basis thus providing better data for the planning and optimisation of E&T programmes.
- Initiatives devoted to improving access to existing research infrastructures for the purpose of education and training.
- Development of new approaches devoted to improving the European mobility of nuclear professionals.
- Strengthening international cooperation with non-EU countries in nuclear E&T and nuclear research.

These developments are described in more detail in the report "Nuclear Education and Training - Key Elements of a Sustainable European Strategy" published by the SNETP in 2010 as well in other, more recent reports published by the OED/NEA and the IAEA .

iii - Challenges

Raising the attractiveness for qualified young people of studies and professions related to the use of nuclear energy remains a key challenge for all stakeholders in nuclear energy.

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Irrespective of the substantial progress achieved by recent initiatives, there remain significant challenges for nuclear education and training. They need to be addressed in order to ensure that lack of qualified human resources will not be a limiting factor for nuclear safety and the continued use of nuclear fission energy.

Not all the priorities recognised during the recent years have, for instance, been sufficiently addressed until now. Some initiatives still need to be completed, and there are also successful past initiatives which are now at risk in a different situation. Thus continued attention is required, for instance, for:

- Cooperation between the different stakeholders in nuclear education and training and continued engagement of nuclear employers in the support of national and international training programmes.
- International information exchange about recent experiences in nuclear education and training in view of identifying best practices and sharing them internationally.
- Regular updating and improving the quality of data providing information about the structure of and the future demand for nuclear professionals in view of supporting anticipated planning and optimisation of nuclear E&T programmes.
- Ensuring easy access to existing research infrastructure for the purpose of education and training.
- Implementation of approaches devoted to achieving Europe-wide recognition of national training achievements in view of increasing the mobility of nuclear professionals.

New challenges derive from the combined impact of the Fukushima accident, the financial crisis and the delays in some new build projects. This combined impact degrades the political and financial perspectives for the use of nuclear energy and thus changes the near- and mid-term perspectives for both the human resources needed and the attractiveness of nuclear professions in competition with other industries for attracting the best young scientists and engineers.

At present, not all potential changes are recognised. Many available data about the standing of education at the European level and the human resources needed still reflect a previous situation where the perspectives of a nuclear renaissance seemed to be obvious. The potential changes may be less relevant for the long-term perspectives, as these generally derive from global conditions such as the need for balanced use of different technologies options in order to ensure the sustainability of energy supply. For the short and medium term, however, changes may by quite significant and need to be addressed by strategies and planning. The relevant data need to be updated and the E&T strategies need to consider them and to reflect increased uncertainties of predictions.

iv - Priorities for the near term

There are a various principles and approaches for nuclear education and training which can help to deal with those changes and to bridge a possible interim with increased uncertainties. The relationship between research on the one hand and education and training on the other hand plays a key role in that respect. The following near-term priorities have been identified:

- Strengthening the links between nuclear fission research and nuclear education and training.
- Continued engagement of the industry and other nuclear employers in the optimisation of research and educational programmes.
- Continued engagement of the European Commission in extending international cooperation with non-EU countries including those engaged in major nuclear programmes.
- Improving the opportunities for young scientists and engineers to work on challenging research topics, e.g. by extending the scope of those programmes.
- Providing opportunities for education and training programmes to include direct experience and experimental work on actual radioactive/irradiated fuel and materials, in synergy with available and future hot lab infrastructure.
- Strengthening the European and international dimension of nuclear education and training programmes thus responding to the increasingly international character of the nuclear industry and to the need to mobilise from different geographical areas qualified human resources for nuclear energy.
- Extending the range of European E&T initiatives from their past focus on higher education to a broader scope covering (vocational) training of technicians and other specialised nuclear workers.

All these actions will contribute to the preservation and dissemination of the acquired knowledge.

To some extent these near-term priorities are already being addressed by recent initiatives[26]. More systematic approaches are under discussion in view of developing solutions more precisely tailored to meet the challenges nuclear education and training is facing in the near future.

26 - The GENTLE project addresses, for instance, some of the priorities listed above.

Conclusions and way forward

The preparation of this revision of the SNETP Strategic Research and Innovation Agenda, has shown that SNETP's long-term vision is still valid, confirming the very important role of nuclear energy for the achievement of SET Plan objectives, and the role of the three R&D pillars to maintain and enhance in the future the sustainability of nuclear energy.

The European electricity generation mix includes a significant share of nuclear energy (about 30%) today and could also in the future: the 2050 Energy Roadmap for example presents as the most efficient, the scenarios with a higher share of nuclear energy. This SRIA develops a long term vision of nuclear fission R&D needs, describing the roadmaps to make nuclear energy more sustainable in the long term, by achieving better usage of natural resources and at the same time reducing the amount and toxicity of final nuclear waste.

Safety being always the first guiding principle for nuclear research, this update of the SRA has emphasised its importance and has made more evident its role for research and innovation. In addition, new challenges for safety have been identified to fully incorporate the first lessons learnt from the Fukushima accident.

The three R&D pillars mantain and enhance the sustainability of nuclear energy

The new SRIA is consistent with the 2020 objectives of the SET Plan, including in NUGENIA the R&D required to maintain the competitiveness of fission technologies together with long term waste management solutions. The SRIA also supports the 2050 vision of the SET Plan, by including in the ESNII priorities the R&D needed to complete the preparation for the demonstration of a new generation of fission reactors with increased sustainability.

SNETP is now mature with its organisation into three technical working groups:

- R&D needs for generation II and III reactors are developed through the NUGENIA roadmaps. NUGENIA with the legal structure of an association is able to organise calls on priority topics in the framework of Private-Private as well as Private-Public Partnerships.
- ESNII is now an official European Industrial Initiative, devoted to developing the fast reactor technologies required for long term sustainability. ESNII has prioritised its demonstration projects: confirming the sodium cooled reactor (ASTRID) as the reference technology, lead cooled systems (MYRRHA and ALFRED) as an alternative technology and a longer term technology with the gas cooled fast reactor (ALLEGRO).
- NC2I is the technical working group reflecting the increased interest of the European energy intensive industry in nuclear cogeneration. NC2I is preparing a concept paper for the development of cogeneration using nuclear energy in order to launch, in the near future, an Industrial Initiative.

The activities of these three main pillars are complemented by dedicated activities and working groups. One example is the identification of the R&D on fuel reprocessing required to obtain the full benefit of closed cycles with fast neutron systems. Another important example is the working group for Education, Training & Knowledge Management, preparing strategies for attracting the most talented people and preserving competence and know-how for future generations.

The next step will consist of the appropriation by the technical working groups of this SRIA in order to define shorter term priorities for its implementation.

Depending on the funding mechanisms put in place in the Horizon 2020 framework, the

Platform will adapt its internal organisation to ensure efficient implementation of the SRIA and in particular to coordinate more widely the R&D programmes of its members, including with the programmes of the corresponding EU Member States.

Funding will remain a major challenge. A significant increase in funding level, compared to the private and public funds at national or European level made available during the past years, will be required to cover properly all the needs identified in the SRIA.

Glossary

ADRIANA	ADvanced Reactor Initiative And Network Arrangement
ADS	Accelerator Driven System
ALLEGRO	GFR demonstration plant
ALFRED	LFR demonstration plant
ASTRID	SFR prototype plant
BWR	Boiling Water reactor
CFD	Computational Fluid Dynamics
CFV	Coeur à Faible effet de Vide de sodium
DBTT	Ductile to Brittle Transition Temperature
DHR	Decay Heat Removal
DGR	Deep underground Geological Repository
EFIT	European Facility for Industrial Transmutation
EIA	Environmental Impact Assessment
ELFR	European LFR
ENEF	European Nuclear Energy Forum
ENSREG	European Nuclear Safety Regulators Group
ESFRI	European Strategy Forum of Research Infrastructures
ESNII	European Sustainable Nuclear Industrial Initiative
ETDR	European Technology Demonstrator Reactor
ETP	European Technology Platform
ETPP	European Technology Pilot Plant
FEED	Front-End Engineering Design
FNR	Fast Neutron Reactor
FP	Fission Products
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum
GUINEVERE	Generator of Uninterrupted Intense NEutrons at the lead VEnus REactor
HLW	High Level Waste
HTR	High Temperature Reactor
HTTR	High Temperature Test Reactor
IAEA	International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IEA	International Energy Agency
IGD-TP	Implementing Geological Disposal Technology Platform
ILW	Intermediate level waste
I&C	Instrumentation and Control
JHR	Jules Horowitz Reactor
LBB	Leak Before Break
LBE	Lead Bismuth eutectic
LFR	Lead-cooled Fast Reactor
LLRF	Low Level RF
LLW	Low level waste
LOCA	Loss of Coolant Accidents
LWR	Light Water Reactor
MA	Minor Actinide
MCCI	Molten Core Concrete Interaction
MOX	Mixed Oxide fuel
MSR	Molten Salt Reactor
MTBF	Mean time between failures
MYRRHA	Multi-purpose hybrid Research Reactor for High-tech Applications
NC2I	Nuclear Cogeneration Industrial Initiative
NEA	Nuclear Energy Agency
NFC	Nuclear Fuel Cycle
NPP	Nuclear Power Plant
NUGENIA	NUclear GENeration II & III Association
NULIFE	Nuclear Plant Life Prediction
NW	Nuclear Waste
OECD	Organisation for Economic Cooperation and Development
PIE	Past Irradiation Examination
PSA	Probabilistic safety assessment
PTS	Pressurised Thermal stock
PWR	Pressurised Water Reactor
RCS	Reactor cooling system
RF	Radio Frequency
RFQ	RF Quadrupole
RPV	Reactor pressure vessel
R&D	Research and Development
SA	Severe Accident
SAMGs	Severe Accident Management Guidelines
SARNET	Severe Accident Research Network
SET	Strategic Energy Technology
SFR	Sodium-cooled Fast Reactor
SNETP	Sustainable Nuclear Energy Technology Platform
SRIA	Strategic Research and Innovation Agenda
SSC	Structures, systems and components
TRISO	Tristructural-isotropic
TRU	Transuranic
VVER	Vodo-Vodyanoi Energetichesky Reactor: Water-Water Power Reactor
WENRA	Western European Nuclear Regulators Association
WGHOF	Working Group of Human and organisational factors
WPFC	Working Party on Scientific Issues of the Fuel Cycle

Contributors

The Chairman of the SRIA Editorial Board 2013, **Dr. Enrique Gonzalez** (CIEMAT) wishes to acknowledge the contribution of the following members of the Platform:

SRIA Editorial Board

- Peter Baeten (SCK•CEN)
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- Anselm Schaefer (ENEN)

Other main contributors

- Alessadro Alemberti (ANSALDO)
- Alexandre Bredimas (LGI)
- Edgar Bogusch (AREVA)
- Marylise Caron-Charles (AREVA)
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- Akos Horvath (MTA-EK)
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- Richard Stainsby (AMEC)
- Jean-Pierre Van Dorsselaere (IRSN)
- Carole Wahide (CEA)
- Lubor Zezula (ÚJV Řež)
- Martin Pecanka (LGI)
- John Sharples (AMEC)
- Ales Laciok (CEZ)
- Karim Ben Ouaghrem (IRSN)
- Abderrahim Al-Mazouzi (EDF)
- Oliver Martin (JRC)
- Liisa Heikinheimo (TVO)

We would also like to express our gratitude to the many other people who contributed to this document directly and in fruitful discussions with those listed above, and more particularly to the EURATOM projects coordinators

Editors : SRIA Editorial Board and SNETP Secretariat | Design: LGI Consulting with Noème

ISBN 978-2-919313-04-4

www.SNETP.eu