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Generation IV reactor systems and fuel cycles (horizon 2030): technological breakthroughs in nuclear fission (int'l RD&DD)

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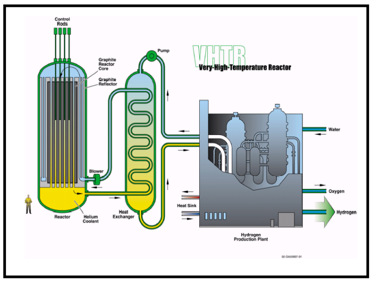
Generation IV International Forum (GIF, 2001)



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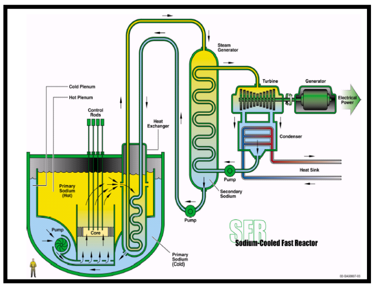

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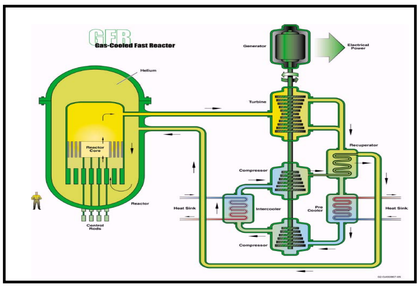

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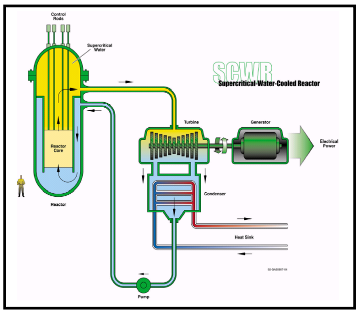

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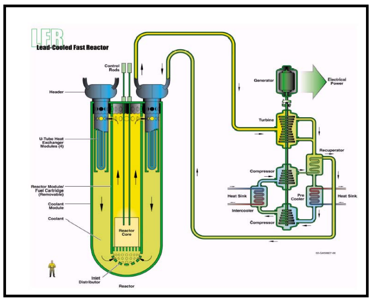

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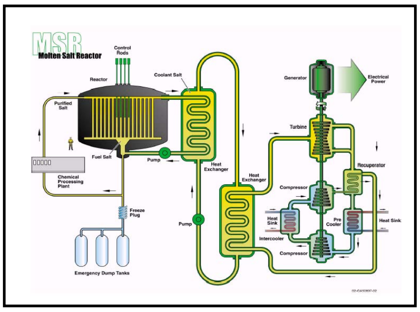


MSR

ABSTRACT

Euratom signed in 2006 the Framework Agreement of the Generation IV International Forum (GIF). As a consequence, all Euratom actions in the area of innovative reactor systems are based on the four "*Technology Goals for industry and society*" set by the GIF:

- sustainability: e.g. enhanced fuel utilisation and optimal waste management
- economics: e.g. minimisation of costs of MWe installed and MWth generated
- safety and reliability : e.g. robust safety architecture, enhanced EUR requirements
- proliferation resistance and physical protection: e.g. impractical separation of Pu.

To set the scene of Generation IV, the history of nuclear fission power is recalled with some discussion about the benefits and drawbacks of each previous Generation:

- Generation I (1950 – 1970): Atoms-for-Peace era plants (4 countries concerned)
- Generation II (1970 - 2000): safety and reliability (30 countries concerned)
- Generation III (2000 - 2030): "evolutionary" steps to further improve safety (EUR)
- Generation IV (horizon after 2030): "visionary" innovation regarding sustainability.

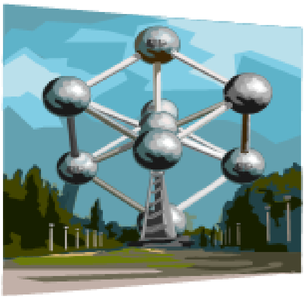
Résumé

Euratom a signé en 2006 l'Accord Cadre du GIF (« *Generation IV International Forum* »). Par conséquent, toutes les actions Euratom dans le domaine des systèmes réacteurs innovants sont basées sur les quatre «Objectifs technologiques pour l'industrie et la société»:

- durabilité : par ex. utilisation des combustibles et gestion optimale des déchets
- économie : par ex. minimisation des coûts du MWe installé et du MWth produit
- sûreté et fiabilité : par ex. solide architecture de sûreté et spécifications très sévères
- résistance à la prolifération et protection physisque : par ex. séparation difficile du Pu.

Pour mettre Generation IV en perspective, on rappelle brièvement l'histoire de la fission nucléaire en discutant les avantages et inconvénients de chaque génération précédente :

- Generation I (1950 – 1970) : centrales nucléaires « *Atoms-for-Peace* » (4 pays)
- Generation II (1970 – 2000) : sûreté et fiabilité (30 pays concernés)
- Generation III (2000 – 2030) : progrès de type "évolutif" pour améliorer la sûreté
- Generation IV (horizon après 2030) : innovation "visionnaire" en durabilité.



« Oui, mes amis, dit Cyrus Smith, je crois qu'un jour l'eau servira de carburant, que l'hydrogène et l'oxygène qui la constituent, utilisés seuls ou ensemble, fourniront une source inépuisable d'énergie et de lumière, d'une intensité dont le charbon n'est pas capable,
et que lorsque les ressources en charbon seront épuisées, nous nous chaufferons grâce à l'eau. L'eau sera le charbon du futur. »
L'île mystérieuse (1874), Jules Verne

Foreword

In this overview paper, the following questions are addressed:

1) *What are the main drivers for innovation in reactor systems and fuel cycles at the horizon 2030 ? (Technology Goals of the "Generation IV International Forum" /GIF)*

2) What kind of response is offered by the international RD&DD programmes in nuclear fission (in particular, Euratom) ? (impact on science, industry and society)

- RD&DD or RD3 for short = Research, technological Development, and engineering Demonstration, industrial Deployment (also called "Innovation Cycle").

The strategy of the EU ("*Sustainable Nuclear Energy Technology Platform*") is discussed in the context of the world-wide collaboration under GIF. The history of Generations I, II and III, is briefly recalled, covering the period 1950 – 2030 (evolutionary steps). The bulk of this paper is on the visionary innovation brought by Generation IV. Four "*GIF Technology Goals for industry and society*" are guiding the international effort, namely: *sustainability*, *economics*, *safety and reliability*, and *proliferation resistance*. Details are given about the specific research plans and the major scientific challenges for each of the six Generation IV reactor systems (expected to be deployed by 2030).

The GIF selection consists of three fast neutron spectrum systems (SFR, GFR, LFR), two thermal/fast systems (SCWR, MSR) and one thermal system (VHTR), i.e.:

- *Very high temperature gas reactors (VHTR)*: cogeneration of high temperature process heat and electricity (efficiency 45 – 50 %) / power 600 MWth or 300 MWe
- *Sodium cooled fast reactors (SFR)*: electricity production and full actinide management (enhanced fuel utilisation) / modules of 50 to plants of 1500 MWe
- *Gas cooled fast reactors (GFR)*: cogeneration of electricity and process heat (enhanced fuel utilisation) / full actinide management / power 1000 MWe
- *Supercritical water cooled reactors (SCWR)*: electricity production at high temperatures (next step in LWR development) / thermal & fast / power 1700 MWe
- *Lead cooled fast reactors (LFR)*: cogeneration of process heat and electricity (full actinide management) / batteries of 10 to plants of 600 MWe / small turnkey plant
- *Molten salt reactors (MSR)*: cogeneration of process heat and electricity (full actinide management) / breeding in thermal (Th) or fast (U – Pu) spectrum.

Conclusions are being drawn, focussing on Europe, as to the role of the stakeholders concerned with innovative reactors and fuel cycles, and the expected impact of Generation IV on science, industry and society (nowadays and in the future "*low-carbon society*").

1 INTRODUCTION / RESEARCH-DEVELOPMENT AND DEMONSTRATION-DEPLOYMENT (RD&DD) IN REACTOR SYSTEMS AND FUEL CYCLES

The globalization that has swept away the barriers to the movement of goods, ideas and people naturally affects the development of research and training. A new approach is required for RD&DD policies at both national and EU level. In this context, the concept of sustainability is a driving force of major importance. The EU definition of sustainability is very close to the one held by Mrs Gro Harlem Brundtland, the former Prime Minister of Norway. This definition was proposed in 1987 at the World Commission on Environment and Development (Brundtland Commission). It describes sustainable development as: "*development that meets the needs of the present without compromising the ability of future generations to meet their own needs*".

Sustainable Development Strategies are implemented in many countries. At the EU level, it is worth recalling the "*Progress Report on the Sustainable Development Strategy* (SDS) 2007" – see EC Communication COM(2007) 642 [1]. The related EU policy has identified seven key sustainability challenges, which will be the subject of annual assessment reports:

1. *Climate Change and Clean Energy* (covering all primary energies, including nuclear)
2. Sustainable Transport
3. Sustainable Consumption and Production
4. Conservation and Management of Natural resources
5. Public Health
6. Social Inclusion, Demography and Migration
7. Global poverty.

The above EU policy relies naturally in part on technological innovation, especially when it comes to *Climate Change and Clean Energy*. In the general debate about innovation and the related RD&DD, there are two types of challenges:

- scientific and technological (S/T) challenges related to Research and technological Development: the main instrument provided by the EU is the Framework Programme
- economic and political (E/P) challenges related to engineering Demonstration and industrial Deployment: the main instruments are economic and regulatory incentives.

The actions of the Research policy in the EU are in line with the objectives of the knowledge-driven society. They are not conducted, however, for the sake of research as a goal *per se*, but as a support to other EU policies, in particular, the Energy policy. This is illustrated in Figure 1 below, in the form of two policy Triangles:

- the "*Knowledge Triangle*" (EU policy for research, innovation and education, with emphasis on RD, that is: Research and Development) - see the FP-7 strategy in the EC Communication of 2005 "*Building the ERA of knowledge for growth*" [2]
  (ERA = "*European Research Area*", launched in the context of the Lisbon Strategy of 2000 - the Lisbon Agenda was revised in 2005 by the EU-25 to give it a clearer focus on a smaller number of policy priorities - "*Working together for growth and jobs*" [3]).

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[1] http://ec.europa.eu/sustainable/docs/com_2007_642_en.pdf

[2] http://ec.europa.eu/research/era/index_en.html

[3] http://europa.eu/scadplus/leg/en/cha/c11325.htm

- the "Energy Triangle" (EU policy for security of supply, competitiveness and sustainable development of energy, with emphasis on DD, that is: Demonstration and Deployment) - see the Energy Package in the EC Communication "*An Energy Policy for Europe*" [4] (EPE, 10 January 2007), subsequently endorsed by the European Council of 8-9 March 2007 [5] (see, in particular, the place of nuclear fission).

The history of nuclear fission in commercial power plants is usually divided into four Generations (called I, II, III and IV), each characterised by S/T and E/P challenges, namely:

- *Generation I (1950 – 1970): Atoms-for-P*eace era plants (originally from naval designs to prototype commercial power plants, only 4 countries concerned worldwide)
- *Generation II (1970 - 2000):* safety and reliability (expansion to 30 countries worldwide), and pioneering steps in sustainability and efficiency (e.g. HTR and FBR)
- *Generation III (2000 - 2030):* evolutionary steps to further improve safety (severe accident management) and competitiveness (e.g. 60 years lifetime, 90 % capacity)
- *Generation IV (horizon after 2030):* visionary innovation regarding sustainability (full actinide management) and competitiveness (e.g. nuclear cogeneration).

At the EU level, nuclear research and training (Euratom programmes) are principally under the responsibility of two Directorates General (DG), using, wherever possible, "*instruments*" from other DGs (in particular, *DG Education and Culture* – see Appendix 1):

- *DG Research* (RTD, located in Brussels), which organises the “indirect actions” of the Seventh Framework Programme [6], i.e. multi-partner projects undertaken by consortia made up of national laboratories, industrial groups and research organisations (both private and public) in the EU, usually on a shared cost basis (50 - 50 % basis)
- *DG Joint Research Centre* [7] (JRC, headquarters in Brussels) which carries out “direct actions” in their own research laboratories (7 scientific institutes in 5 Member States). Nuclear research at DG JRC focuses on innovative nuclear fuels and materials, neutron measurements, non-proliferation issues (safeguards) and numerical modelling. The JRC has been designated in 2006 as "*Implementing Agent*" of the GIF Framework Agreement, and will represent all interested EU Member States at GIF (see Section 6).

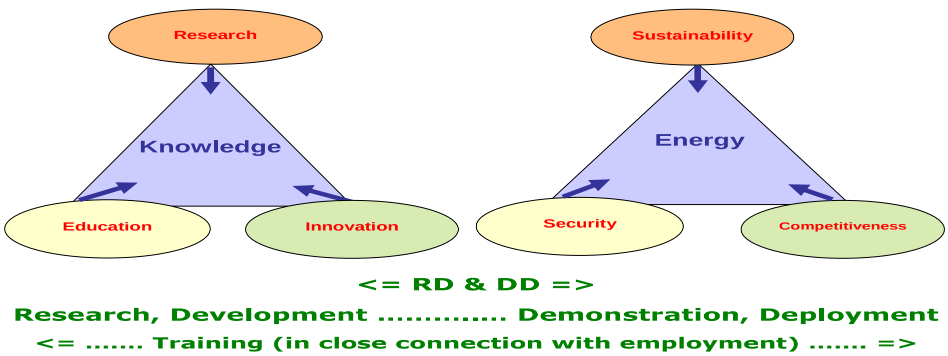

Figure 1 – Innovation Cycle (RD&DD) / "*Knowledge Triangle*" (Research and Development) and "*Energy Triangle*" (Demonstration and Deployment) + Training

[4] http://ec.europa.eu/energy/energy_policy/index_en.htm

[5] http://www.consilium.europa.eu/ueDocs/cms_Data/docs/pressData/en/ec/93135.pdf

[6] http://cordis.europa.eu/FP-7/euratom/home_en.html

[7] http://www.jrc.cec.eu.int/

2 GLOBAL COOPERATION AMONGST ALL NUCLEAR STAKEHOLDERS (CRITICAL YEAR 2012) / RESEARCH, SYSTEMS SUPPLIERS, ENERGY PROVIDERS, REGULATORY BODIES, UNIVERSITIES AND CIVIL SOCIETY

At the above-mentioned March 2007 European Council, the role of nuclear energy was one of the main debating points. The Presidency Conclusions reiterated the established position that it is for each EU nation to decide whether to use nuclear power. The EU Energy Ministers actually reaffirmed that the Energy Policy for Europe (EPE) should contribute in a balanced way to the following three objectives:

- *increasing security of supply* (nicknamed "Moscow")
- *ensuring the competitiveness of the European energy industry so as to provide energy at the best possible prices for citizens and companies* (nicknamed "Lisbon")
- *promoting environmental sustainability (with emphasis on the objective to limit the rise in global temperatures to 2°C by 2050)* (nicknamed "Kyoto").

At the February 2008 (Energy) European Council [8], another important step was taken regarding nuclear fission at the EU level. Six priority *Industrial Initiatives* were launched. They should be of a voluntary nature, and can take the form of public-private partnerships or of joint programming by groups of interested Member States. Proposals should demonstrate cost-effectiveness and added value. These initiatives are: "European wind initiative", "Solar Europe initiative", "Bio-Energy Europe initiative", "European CO_2 capture, transport and sequestration initiative", "European electricity grid initiative" and "*Nuclear fission initiative*".

As far as research and development in nuclear fission is concerned, discussions on a long-term strategy started in the Michelangelo network which became a Euratom FP-5 (1998 – 2002) project (MICANET) and continued in three Euratom FP-6 (2003 – 2006) projects:

- *SNF-TP (Sustainable Nuclear Fission, 2 years, from October 2006, coordinator CEA)*
- *PATEROS (Partitioning and Transmutation European Roadmap for Sustainable nuclear energy, 2 years, from September 2006, coordinator SCK-CEN)*
- *CARD (Coordination of research, development and demonstration priorities and strategies for geological disposal, 1 year, from September 2006, coordinator NIREX).*

All major nuclear stakeholders participate in these strategic discussions, i.e.:

- research organisations (public and private, power and medical applications)
- systems suppliers (e.g. nuclear vendors, engineering companies, etc)
- energy providers (e.g. electric utilities, heat and/or hydrogen vendors, etc)
- regulatory bodies and associated technical safety organisations (TSO)
- education and training (E&T) institutions, and, in particular, universities
- civil society and the international institutional framework (IAEA and OECD/NEA).

As a result of the above Euratom FP-5 and FP-6 projects, a European Technology Platform (ETP) was launched in 2007 to bring together all the above major stakeholders in the EU and discuss a common strategy on research and development (RD) in nuclear fission:

[8] http://register.consilium.europa.eu/pdf/en/08/st06/st06326.en08.pdf

(1) Sustainable Nuclear Energy Technology Platform (SNE-TP [9])
- launch event in Brussels on 21 September 2007, platform composed principally of research organisations from both the public and private sectors
- the SNE-TP consists of a General Assembly (biennial meetings), a Governing Board and an Executive Committee that supervises 3 main activities:
  * *Strategic Research Agenda* (SRA)
  * *Deployment Strategy* (DS), including policy framework
  * Knowledge Management and Education & Training (coordinated by ENEN).

To launch the SNE-TP, a "*Vision Report*" was published, signed by 35 EU organisations interested in nuclear fission research (including 11 from industry). It proposes a vision for the near, medium and long-term development of nuclear fission energy technologies, with the aim of achieving a sustainable production of nuclear energy, significant progress in economic performance and the highest level of safety, as well as resistance to proliferation.

In the *Preliminary research roadmaps for the different technologies,* proposed in this Report, the year 2012 is very critical. Here is an excerpt, focusing on innovative systems:

- *Generation II – III LWRs* / 2012: Viability of SCWR
- *Generation IV fast-neutron reactors (sustainability)*:
  - Sodium-cooled Fast Reactor (SFR)
  2012: Confirmation of design options / 2020: Start-up of operations
  - Gas- and Lead-cooled Fast Reactors, as well as Accelerator-Driven Systems:
  2012: Selection of a second type of fast-neutron system of importance for Europe.
- *New applications of nuclear energy* / Very High-temperature Reator:
  2012: Confirmation of key technologies / 2015 - 2020: Construction of a VHTR and demonstration of cogeneration applications (alternative fuels to oil)
- *Advanced recycling processes* / 2012: Selection of technologies for the closed fuel cycle with the development of MA bearing fuels / 2012 - 2017: construction of a fuel-manufacturing workshop and a micro-pilot plant for minor actinide recycling.

In 2008, the SNE-TP discussions focus on the *Strategic Research Agenda* and *Industrial Initiative on Nuclear Fission.* The aim is to maintain the EU's leadership in the nuclear energy sector. In addition to the SNE-TP that focuses on research and development (RD), two other EU actions were launched, focussing on demonstration and deployment (DD):

(2) European Nuclear Energy Forum (ENEF [10])
- launch event in Bratislava on 26 November 2007, forum composed principally of industrial stakeholders and non-governmental organisations (NGOs)
- the ENEF consists of three Working Groups:
  * *Opportunities*
  * *Risks* of nuclear energy, including education and training, which is led by EoN
  * *Information* and transparency issues.

(3) High Level Group (HLG)
- launch event in Brussels on 12 October 2007, group composed principally of senior officials from national regulatory or nuclear safety authorities.

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[9] www.snetp.eu

[10] http://ec.europa.eu/energy/nuclear/forum/index_en.htm

Global partnership to support the "nuclear Renaissance" (GIF, GNEP, GNI-IFC)

Some signals of a “nuclear Renaissance” (or re-deployment) can be observed in the EU, mainly in countries that devoted special efforts to improving the economics and the sustainability of the nuclear option (in their power plants and related fuel cycles). This has been recently demonstrated, in particular, in Finland and France who were the first countries to take firm decisions about the back-end of the nuclear life cycle prior to decide the construction of new NPPs.

It is worth recalling the role of the “*European Utility Requirement*” [11]/EUR/ association (created in 1991) in this potential re-deployment. The EUR association includes 12 utilities, namely: British Energy (UK), Suez – Tractebel (BE), Electricité de France (FR), NRG (NL), IBERDROLA (ES), VGB PowerTech (DE), SOGIN (IT), Vattenfall (SE), TVO and FORTUM (FI), Swissnuclear (CH) and Rosenergoatom (RF). They bring together the major electricity producers in Europe with the aim to harmonise design targets and to fix the technical specifications for evolutionary LWRs, that is: Generation III. Following a rather prescriptive approach, a total of approximately 4 000 individual requirements were fixed, that deal with all the topics a utility has to address to have a LWR developed and built in Europe. A benchmark comparison has been made with the similar US document EPRI/URD.

As of today, only Finland and Sweden have achieved a national consensus regarding the back-end of the nuclear life cycle. They took firm decisions about an underground disposal site for high-level waste or spent fuel. They are actually the only countries in the world whose governments agreed on a plan and a budget for a repository for spent fuel. In Finland, the decision regarding spent fuel (high activity) was taken in 2001 (build of a reversible geological disposal in granite near Olkiluoto). The site should open around 2015: an underground laboratory *Onkalo* is under construction. In Sweden, the spent fuel site selection has almost been completed for a reversible geological disposal in granite. Target date for operation is also 2015: an underground laboratory has been operating in Aspö since 1994.

It is no wonder that the construction of the first Generation III reactor started in Finland. Following a public debate (approval of Finnish Parliament in May 2002 and then by the government), the power company TVO ordered the construction of the first EPR (1600 MWe, Olkiluoto, unit 3): a contract was signed with AREVA NP and Siemens PG consortium in December 2003 (commercial operation originally planned for 2009).

France is on the way to achieve a national consensus on the management of the back-end of the fuel cycle. Their approach is summarized in a bill voted on by the National Assembly in 2006, named *"Programme Act on the Sustainable Management of Radioactive Materials and Wastes"* [12]. The text establishes a national programme (with milestones) to build a deep-geologic repository for the disposal of retrievable radioactive waste while confirming the need to keep conducting research on waste management. Research is being conducted on a reversible disposal in deep geological formations. An application for the authorisation of a site and a design can be made in 2015 and the centre can be in operation in 2025.

Also following a public debate, the French utility EDF was authorized in April 2007 to build the first EPR unit (1600 MWe, Flamanville, unit 3, operation planned for 2012). The

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[11] http://www.europeanutilityrequirements.org/

[12] http://www.andra.fr/IMG/pdf/060621_loi-en.pdf

total investment cost is EUR 3.3 billion (60 % for nuclear and 40 % for conventional). EDF envisages building 1 to 1.5 EPRs of 1600 MWe per year, starting from 2020 for 20 years.

The "nuclear Renaissance" is actually more tangible outside the EU. In 2007 worldwide, half of the new 34 nuclear reactors being built are in developing (usually "emerging") countries. The demand for civilian nuclear power by developing countries is rising sharply, ending the monopoly by industrialized countries over nuclear energy. Three strong factors are driving up the global interest in nuclear energy (similar to the above-mentioned EPE):

- increasing concerns about energy security
- the steady growth of energy demand (under affordable conditions)
- the challenge of climate change.

The "nuclear Renaissance" focuses naturally on Generation III. Certain countries, however, are already looking at the longer term, that is: Generation IV. Announcements and roadmaps to build systems of the Generation IV type were issued in France and the USA.

At New Year 2006, i.e. a few months before the French vote on the above bill on the Sustainable Management of Radioactive Wastes, President Jacques Chirac addressed the business leaders and unions, announcing that he had "*decided to immediately launch work by the French Atomic Energy Commission (CEA) on a prototype fourth-generation reactor, to go into service in 2020* [13]". With the construction of a prototype Generation IV nuclear reactor, France is determined to remain a world leader in nuclear energy. No specific reactor type was mentioned, however. At the time being, the CEA is focusing its efforts on the fast spectrum reactors (sodium SFR and helium cooled GFR), whereas the industrial partner (systems supplier AREVA) is focusing on the thermal spectrum high-temperature reactor V/HTR. The holistic approach of the French nuclear policy is quite remarkable: the announcement about Generation IV and the bill about sustainable waste management go naturally hand in hand.

Also in 2006, the USA (President G. W. Bush) announced the *Global Nuclear Energy Partnership* (GNEP, 2006). This US initiative is, in fact, a part of their Advanced Energy Initiative that was first announced in January 2006 (after a one-year discussion with the UK, France, Russia, China and Japan – plus the UK and the IAEA as observers) and then followed by a Strategic Plan released on 10 January 2007 by the US Department of Energy [14]. GNEP seeks to develop worldwide consensus on enabling expanded use of clean, safe, and affordable nuclear energy to meet growing electricity demand. GNEP proposes a nuclear fuel cycle that enhances energy security, while promoting non-proliferation. For short, the GNEP proposes to divide the world in two categories: the "*fuel suppliers*" on one side and the "*fuel users*" on the other side. In 2007 the US Energy Department called for the construction of a "*Nuclear Fuel Recycling Center*," for reprocessing and fuel fabrication, and of an "*Advanced Recycling Reactor*" as a prototype for 40-75 fast burner reactors. It is thus the end of the Jimmy Carter doctrine of non-reprocessing (US President, 1977 – 1981), launched in 1977.

The Russian Federation (President V. Putin) launched a similar scheme, termed *Global Nuclear Infrastructure - International Fuel Centre* (GNI-IFC, 2006). Both initiatives, GNEP and GNI-IFC, are attempts to limit access to sensitive technologies (that is: fuel fabrication and reprocessing) to a limited number of "*fuel supplier*" countries and to further tighten up the international safeguards regime in the context of an expanded use of nuclear energy.

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[13] http://www.diplomatie.gouv.fr/actu/bulletin.asp?liste=20060215.html

[14] http://www.gnep.energy.gov/

Under the umbrella of GNEP and GNI-IFC, international collaboration is also stimulated on the regulatory side. A number of national regulatory authorities agreed to share the resources and the knowledge accumulated during their assessment of new reactor designs. As a result, the *Multinational Design Evaluation Program* [15] (MDEP) was launched in 2005 by the NRC. As of March 2008, 10 countries participate in this programme: Canada, Finland, France, Japan, the Republic of Korea, the United Kingdom and the United States, plus China, the Russian Federation and South Africa.

The objective of the MDEP is to "*assure a harmonized approach to long-term safety, risk, and regulatory issues in the development of next generation systems*". Therefore an evaluation methodology has been developed for safety goals, reliability of passive safety systems, the approach to severe accidents and quality assurance standards. Both safety and economics have been improved by using a "risk-informed" approach during the entire design process, as a complement to the deterministic approach (mastering the uncertainties, e.g. using PSA). Special attention is devoted to technology-neutral risk and safety criteria, based on an innovative approach using new instruments such as *Objective Provision Trees* (OPT) and *Lines of Protection* (LOP). In this way, safety and non-proliferation objectives also become more transparent at the international level.

Three stages are proposed in the MDEP:

- **stage 1** is under way: focus on the planned design reviews associated with the AREVA EPR reactor in collaboration with the US, Finnish and French Safety Authorities.

- **stage 2** is intended to be more extensive: identify common regulatory practices and regulations that enhance safety (in particular, in the areas of design basis accidents and emergency core cooling system performances, severe accident requirements, digital I&C) and foster convergence of codes and standards for designs across the world (manufacturing and quality assurance) – the technical secretariat is with OECD/NEA

- **stage 3** (implementation and expansion): uses the results of the stage 2 effort to review the advanced reactor designs of Generation IV reactors. Special attention is devoted to those countries with less experienced and extensive regulatory regimes, by encouraging a comprehensive safety.

It is worth recalling the IAEA definition of advanced nuclear plant designs:

- "*evolutionary*" (Generation III): the designs emphasise improvements based on proven technology and experience. No prototype is needed for their industrial deployment. From a safety point of view, the two aims of "evolutionary" reactors are a further reduction of the core damage frequency (e.g. increased use of passive safety features, wherever justified) and a limitation of the off-site consequences, should a severe accident occur (e.g. strengthening the function of the containment)

- *"visionary" or "revolutionary"* (Generation IV): the designs emphasise the use of new or entirely revisited features, particularly with regard to full actinide management and enhanced safety. Prototypes will be needed for their industrial deployment. The main aim of these reactors is to integrate all four GIF Technology Goals in the design ("built in" features, not "added") and, in particular, to develop a "robust" safety architecture to demonstrate the "practical elimination" of severe accidents.

[15] http://www.nea.fr/html/general/press/2008/2008-01.html

3 GENERATIONS I, II AND III / A CONTINUUM OF NUCLEAR TECHNOLOGY DEVELOPMENTS FROM 1950 TO 2030 ("EVOLUTIONARY" STEPS)

Generation I (1950 – 1970 / power of 50 – 500 MWe / USA, Soviet Union, France, UK)

These were the prototype commercial reactors of the 1950s and 1960s – as a result of the *Atoms-for-Peace* initiative - in the USA, in the former Soviet Union (USSR), in France and in the United Kingdom. In 1953, US President Dwight D. Eisenhower made a famous address in response to the escalating nuclear arms race between the United States and the Soviet Union ("*Atoms for peace*", UN General Assembly, New York, 8 December 1953).

Generation I mostly used natural uranium fuel (to avoid the need for enrichment), used graphite (or heavy water) as a moderator and pressurized CO_2 as a coolant. Relatively few Generation I reactors are still running (e.g. in the EU, two *MAGNOX* stations in the UK, due to be closed by 2010 - out of the original eleven - *MAGnesium Non-OXidising,* alloy of magnesium with small amounts of aluminium).

The period 1950 – 1970 was actually very rich in new reactor concepts. Here is a short history of power generation by nuclear fission:

- 1951: in the USA, the Experimental Breeder Reactor-1 EBR-1 provided electricity for "the Four Light Bulbs" (Idaho National Energy and Environmental Laboratory)
- 1954: in the Soviet Union, the world's first nuclear power plant generated electricity in Obninsk, a water-cooled and graphite-moderated, with a design capacity of 30 MWth or 5 MWe (actually a predecessor of the RBMK, dubbed "Atom Mirny", that is: "peaceful atom", Minatom), closed in 2002
- 1954: in the USA, the world's first nuclear-powered submarine, *USS Nautilus*, was launched (a PWR of 10 MWth / legendary Admiral H. Rickover, father of the nuclear navy) - in 1959 both USA and USSR launched their first nuclear-powered surface vessels
- then came a series of large prototype reactors of large-scale commercial nuclear power plants (NPPs): Gas-cooled Reactor *Magnox* (1956, Calder Hall, 50 MWe) in the UK; light-water reactors of Westinghouse *PWR* (1957, Shippingport, 60 MWe, owned by US-DOE) and of General Electric *BWR* (1959, Dresden, first one privately built) in the USA; Gas-cooled Reactor (1963, UNGG, Chinon A1, 70 MWe) in France; and *RBMK* (100 MWe prototype in 1963) in the former Soviet Union.

It is worth recalling that the first PWR outside the USA was the *BR3* (Belgian Reactor n° 3, first chain reaction in 1962, SCK-CEN Mol): it was used as a prototype for the construction and operation of later commercial PWRs.

Generation II (1970 – 2000 / power of 500 – 1300 MWe / 30 countries worldwide)

These are the commercial reactors, deployed since the 1970s, as a result of the first fossil energy crisis (OPEC oil crisis of 1974). The large majority are still in operation worldwide today: they are derived from designs originally developed for naval use. In 2007, about 85% of the world's nuclear electricity was generated by reactors of Generation II.

The Generation II reactors typically use enriched uranium fuel and are mostly cooled and moderated by water. They include such light-water reactors (LWR) as the boiling water reactor (*BWR*) and the pressurized water reactor (*PWR*). The fuel, ceramic uranium dioxide UO_2, is typically encased in long zirconium alloy tubes. The uranium-235 is enriched from its original 0.7 % abundance to 3.5 – 5.0 %. In the UK, the second generation of reactors are advanced gas-cooled reactors (*AGR*). In Canada and Romania, they are the *CANDU* reactors (CANadian Deuterium Uranium heavy-water moderated and natural uranium fuelled). In the Russian Federation, they are the VVERs (*Vodaa Vodiannee Energititscherski Reactor)* of types V-213 and V-1000 and the RBMKs *(Reactor Bolshoï Moshchnosti Kalani,* also called LWGR, i.e. boiling Light-Water cooled, Graphite-moderated, pressure-tube Reactors*)* of the second generation.

The pioneering period 1950 – 1970 (Generation I) also saw the first developments of innovative non-LWR designs, which became prototype power reactors in the next period 1970 – 2030 (Generations II and III) and can even be considered as precursors of most Generation IV reactors. Very innovative improvements were achieved in two areas:

(1) better utilisation of natural resources (it was thought that fresh uranium was scarce and that plutonium could be bred quite easily): conversion of fertile into fissile isotopes, in particular, through the breeding cycle U-238 => Pu-239 in FBRs (start in 1963, USA):

(2) enhanced thermodynamic efficiency and utilisation of natural resources (it was thought that nuclear cogeneration of high enthalpy heat was needed for industry): higher core outlet temperatures and breeding cycle Th-232 => U-233 in HTRs (start in 1967, Europe).

(1) Better utilisation of natural resources in SFRs
(breeding of fertile uranium U-238 in fast neutron spectrum reactors)

Fertile uranium U-238 is converted to fissile plutonium Pu-239 during reactor operation. The fast breeder reactors (FBRs) are aiming at a better utilisation of natural resources (conversion fertile - fissile), thereby improving the nuclear waste management process. A number of (sodium cooled) FBRs were constructed (MOX fuel in pellets / in pin), such as:

- in the USA: demonstration reactor Enrico Fermi (60 MWe) commissioned in 1963 and Fast Flux Test Facility (FFTF, 400 MW) in 1970 – followed later on by the Advanced Liquid Metal Reactor (ALMR) programme in the late 1980s (including the Integral Fast Reactor /IFR/ operated until 1994 at EBR II by Argonne National Laboratory - West and the PRISM by General Electric)
- in the former Soviet Union: experimental reactor BOR-60 (Dimitrovgrad) commissioned in 1969, followed by BN-350 (Aktau, Kazakhstan) in 1973 and BN-600 (Beloyarsk) in 1980 – in addition to sodium, a long experience is also available in lead-bismuth Pb-Bi for submarines (since the 1950s)
- in France: experimental reactor *Rapsodie* (40 MWth) commissioned in 1967, followed by *PHENIX* (250 MWe, started in 1973, stopped in 1998 and restarted in 2003) and *SUPERPHENIX* (1240 MWe, started in 1985 and closed in 1997)
- in the United Kingdom: PFR (250 MWe) in Dounray (Scotland) started in 1974
- in Germany, SNR-300 (completed in 1985 but never operational, closed in 1991).

Nowadays, the only fast flux facilities available (Figure 9) are Phenix in France (due to be shut-down in 2009), and BOR-60 and BN-600 in the Russian Federation. As far as Japan is concerned, JOYO (140 MWth, commissioned in 1978) is temporarily closed and MONJU (280 MWe, commissioned in 1994) has been in long-term shut-down (sodium leak accident in 1995) but is planned to be back in operation in October 2008. Worth mentioning also is the Indian programme: their experimental reactor *FBTR* (40 MWth) was commissioned in 1985.

(2) Enhanced thermodynamic efficiency and utilisation of natural resources in HTRs (higher core temperatures and breeding of fertile thorium in thermal reactors)

A number of high-temperature gas reactors (HTR) were constructed during the late 1960s (usually helium-cooled). The early history of the thorium cycle is discussed in Appendix 2.

Nowadays, several HTR projects of industrial interest are being developed by engineering companies, with emphasis on the enhanced thermodynamic efficiency, such as:

- PBMR in the Republic of South Africa (*Pebble Bed Modular Reactor*) designed by PBMR Ltd and Westinghouse, demo plant of 115 MWe planned to operate by 2012
- GT-HTR300C (*Gas Turbine HTR)* in Japan
- ANTARES [16] (*A New Technology with Advanced Reactor for Energy Supply*) in France (AREVA NP, cogeneration of heat up to 800°C / max 600 MWth or 300 MWe)
- NHDD (*Nuclear Hydrogen Development and Demonstration*) in South Korea
- GT-MHR[17] (*Gas Turbine Modular Helium Reactor*) in the Russian Federation / 285 MWe / planned to be built in Seversk/Tomsk-7, as part of the Russian plutonium disposition strategy supported by US-DOE)
- NGNP [18] (*Next Generation Nuclear Plant*, actually a Generation IV demo plant), designed to provide large quantities of process heat and hydrogen, operation planned in the USA by 2018 (Westinghouse, General Atomics, AREVA NP Inc, PBMR Ltd).

A number of experimental reactors are also operating, such as HTTR (Japan, JAEA, 30 MWth) and HTR-10 (China, INET, 10 MWth) to support the development of advanced concepts, and the cogeneration of electricity and hydrogen, or other nuclear heat applications.

Note: global statistics of nuclear fission technologies for electricity and propulsion

At the end of 2007, the world fleet consisted of 439 commercial NPPs, including 146 reactors in 15 of the 27 EU Member States (Figure 2 below). Their distribution in terms of technology is as follows [19]: 215 PWR and 50 VVER (or LWGR, that is: Russian design PWR), 94 BWR, 44 PHWR (Pressurized Heavy Water Reactors, e.g. CANDU), 18 Gas-cooled Reactors (including 14 AGR and 2 Magnox reactors), 16 RBMK (Russian design), 2 Fast Breeder Reactors (FBR, namely: the Russian BN-600 and the French Phenix).

It should be recalled, however, that growth in the USA's nuclear energy industry has been at a standstill for the last 30 years since the severe accident at Pennsylvania's Three Mile Island nuclear plant in 1979 (a PWR-900 of Generation II), a status that was reinforced with the meltdown at the Chernobyl plant in Ukraine in 1986 (a RBMK-1000 of Generation I).

In addition, it is generally estimated that approximately 220 reactors provide power to 150 ships and submarines worldwide. Nuclear power is particularly suitable for vessels which need to be at sea for long periods without refuelling (e.g. aircraft carriers and submarine propulsion). The current technology for nuclear propulsion (electric engine) is predominantly PWR. Reactor power ranges from 10 MWth (in prototypes) up to 200 MWth in the larger submarines and 300 MWth in surface ships such as the Russian *Kirov* class battle cruisers. This technology has proven to be robust for submarine propulsion, however the size, weight and costs of this technology are currently prohibitive for merchant ship propulsion.

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[16] http://www.areva-np.com/common/liblocal/docs/product_sheet/2/4_voletsVHTR2005.pdf

[17] http://gt-mhr.ga.com/R_GTMHR_Project_all.html

[18] http://nuclear.inl.gov/deliverables/

[19] http://www.iaea.org/programmes/a2/

Finally, worldwide 56 countries operate a total of 284 research reactors for scientific purposes and the production of medical (for imaging and therapy) and industrial isotopes.

Generation III ("evolutionary" / 2000 - 2030 / from modules of 150 MWe to large reactors of 1600 MWe)

These are the reactors designed in the 1990s: the first ones are in operation in East-Asia and others are under construction. Referred to as advanced NPPs, these reactors include:

- large power reactors, such as the *European (or Evolutionary) Pressurized power Reactor* (EPR) and the Russian VVER-1200 (AES-2006) as well as the US / Japanese *Advanced Boiling* (ABWR) and *Advanced Pressurized Water Reactors* (APWR)
- small power modules, such as the above-mentioned *Pebble Bed Modular Reactor* (PBMR Ltd, 115 MWe) and the *International Reactor Innovative and Secure* (integral compact design IRIS, 335 MWe, Westinghouse, to be commercialised by 2015).

They are developments of Generation II with enhanced safety and performance characteristics. Some of these reactors have very advanced technological features (such as a simpler design and passive safety systems) that may foreshadow Generation IV reactors.

In the EU, the "nuclear Renaissance" is tangible especially in Finland (Olkiluoto 3, first concrete pouring in 2005) and in France (Flamanville 3, first concrete pouring in 2007). Other positive announcements were made. For example, the Finnish power companies TVO (Olkiluoto) and FORTUM (Loviisa) also announced in March 2007 their intention to build two additional units. In the Czech Republic, the government foresees the construction of two reactors, probably at Temelin, to replace capacity at Dukovany after 2020. As far as the Baltic region is concerned, Lithuania in collaboration with Estonia, Latvia and Poland signed a Memorandum of Understanding (MoU) in March 2006, aimed at preparing the new Ignalina project. In the Netherlands, the government intends to review national laws and regulations related to the construction of new NPPs. The UK is currently debating its energy policy for the next 25 years and is considering replacing its ageing nuclear park. Other EU countries, including Slovakia, Bulgaria and Romania, are re-considering nuclear power by either extending the rating and operating life of existing plants or planning new build, most likely of the Generation III type.

Worldwide at the end of 2007, a total of 34 new nuclear stations were under construction, spread over 14 countries with a total power of more than 27 GWe, that is: 8 % of the existing capacity (370 GWe). They are a mix of Generation II (primarily) and Generation III types, namely: 19 in Asia (6 in India, 5 in China and 2 in Taiwan, 1 in Japan, 3 in the Republic of Korea, 1 in Pakistan and 1 in Iran); 7 in Russia; 4 in the EU (2 in Bulgaria; 1 in Finland, 1 in France); 2 in Ukraine, 1 in the USA and 1 in Argentina. In terms of technology, the distribution is as follows: 26 PWR (including the two first EPR in Finland and France, and 11 VVER), 2 advanced BWR, 4 PHWR, 1 RBMK and 1 FBR (the two latter in Russia: RBMK of Generation III in Kursk-5 and advanced BN-800 in Beloyarsk-4).

Finally, worldwide, more than 35 commercial power reactors were firmly planned (representing 10 % of the existing capacity), namely: 11 PWR, 10 VVER, 9 BWR, 2 PHWR, 2 FBR (in Russia) and another one. Finally another 26 power reactors are proposed (that is: site and reactor type known), including 10 in the USA. Also worth mentioning is that a total of 77 countries are exploring nuclear energy as a potential option and have introduced requests to the IAEA. The vast majority of these countries do not operate their own facilities for the enrichment of uranium, instead relying on the purchase of nuclear fuel (usually from nuclear weapons states by the terms of the Non-Proliferation Treaty).

Worldwide in industrial countries, all relevant signs indicate that, at least for some time (until circa 2030), advanced nuclear plants will be of the evolutionary type (i.e. Generation III), that is: developments from the light and heavy water cooled and moderated plants that are the principal types in use today. As a consequence, in the 21st century, several generations of reactors (II, III and IV) will most likely coexist (together with fusion energy reactors).

Note: reminder about fissile, fertile and the concept of breeder reactors

"*Fissile*" material is material for which fission is possible with neutrons that have low kinetic energy. Remember that thermal reactors use slow incident neutrons in the range 0.01 – 1 eV (thermal) and around 1 eV (epithermal). Fast reactors use fast incident neutrons in the range 10 keV – 25 MeV. "*Fissionable*" material is material for which fission caused by neutron absorption is possible provided the kinetic energy + the binding energy is greater than the critical energy. Notably, uranium-238 is fissionable but not fissile. Examples of fissile nuclides are U-233, U-235, Pu-239, and Pu-241. In general, actinide isotopes with an odd number of neutrons are fissile. A "*fertile*" nuclide is one which leads to the production of a fissile nuclide, on absorption of a neutron. Th-232 and U-238 are well-known fertile nuclides.

Nuclear transmutations take place in a reactor due to continual nuclear interactions and radioactive decays, leading to the production of a variety of new nuclei, not present in the initial feed of materials in the reactor. Some of the newly produced nuclei are advantageous and some disadvantageous. The advantageous new nuclei, produced in the transmutations, include "fissile" material. The term "breeding" refers to production of more fissile than consumed. Through a proper combination of fissile (U-235, U-233 or Pu-239), fertile (U-238 or Th-232) and other materials arranged in a carefully worked out geometry, together with proper reprocessing schemes, it is possible to realize a fissile production rate that exceeds the fissile consumption rate. A reactor system in which this is realized is called a "breeder". Breeding refers thus to the artificial production of fuel for nuclear reactors.

The degree of conversion of fertile to fissile is given by the conversion ratio (CR), defined as *CR = Fissile mass produced / Fissile mass destroyed.* The specific index CR is called *breeding ratio (BR)* if it exceeds unity, indicating production in excess of consumption. The *breeding gain G = BR-1, that is: Fissile gained / Fissile destroyed*.

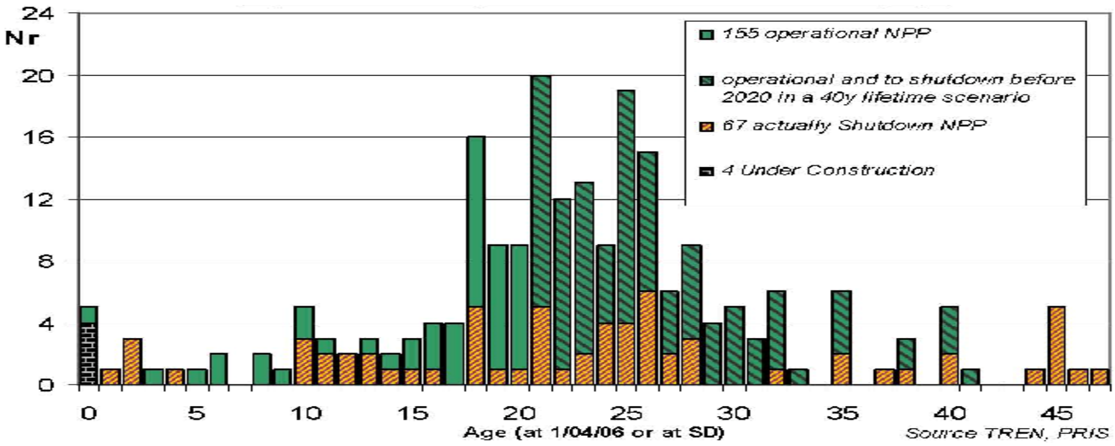

Figure 2 – Number and age of NPPs in the EU-27 in 2006 (155 units in 15 Member States, to be compared with the historical high record of 175 units in 1989)

4 MAIN DRIVER FOR INNOVATION IN REACTOR SYSTEMS AND FUEL CYCLES: SUSTAINABILITY (GIF TECHNOLOGY GOAL N° 1)

There is no doubt that the Generation II nuclear power units of Western design have been found to be safe and reliable: period 1970 – 2000 / power of 500 – 1300 MWe / 30 countries worldwide. Further improvements, however, are possible, in particular, in the areas of reactor safety and reliability (e.g. severe accident management) and enhanced fuel utilisation (e.g. MOX and high burnup). This is the main objective of the Generation III reactors: "*evolutionary*" developments / period 2000 - 2030 / modules of 150 MWe – large reactors of 1600 MWe. "*Visionary*" or "*revolutionary*" improvements are being investigated in Generation IV: horizon after 2030 / modules (batteries) of 10 MWe – large reactors of 1700 MWe, with emphasis on sustainability and economics (e.g. nuclear cogeneration of heat and power /CHP/). Those very innovative improvements will also require new types of fuel cycles that are fully closed in terms of actinide management (e.g. Th, Pa, U, Np, Pu, Am, Cm).

Nuclear cogeneration of heat, usually without GHG emissions, seems very appropriate for a number of processes needing very high, high, medium and/or low temperature heat, in addition to electricity (see Figure 3 in Section 6), that is:

- very high-temperature process heat (800 – 1200 °C)

- e.g. very efficient electricity generation (using gas turbines), iron manufacturing (direct reduction methods), cement manufacturing

- high-temperature process heat (550 – 800 °C):

- e.g. production of hydrogen by SI (thermo-chemical cycle of Sulfuric acid - Iodine process) or by SMR (Steam Methane Reforming)
- hydrogen for bitumen processing, refinery applications, pipeline hydrogen

- medium temperature process heat (250 – 500 °C):

- e.g. coal liquefaction (coal to liquid /CTL/ without CO_2 emission) / gasification (syngas from natural gas); oil refining and desulphurisation of heavy oils (e.g. oil sands recovery and processing in Alberta, Canada); biomass

- low temperature process heat (30 - 250 °C):

- e.g. steam for district heating and seawater desalination for water-scarce countries.

Generation IV will probably be deployed after 2030. The Generation IV systems will be very innovative, especially regarding the following aspects (the four so-called "*Technology Goals for industry and society*", proposed in the GIF roadmap of 2002 – Section 6):

- sustainability: enhanced fuel utilisation and optimal waste management
- economics: minimisation of costs of MWe installed and MWth generated
- safety and reliability : robust safety architecture and enhanced EUR requirements
- proliferation resistance and physical protection: impractical separation of plutonium.

Major strategies for the back-end of the fuel cycle
(recycling or once-through / key question: *is plutonium an asset or a liability ?*)

To understand the complex problem of the back-end of the fuel cycle, it is necessary to define exactly what is meant by the rather general term "nuclear waste". Nuclear waste consists of a wide range of radioactive elements which are defined in terms of their activity levels – low- and medium-level waste (LMLW) and high-level waste (HLW) – and the duration of this activity. The high-level waste, in particular, contains more than 90 % of the radioactivity and accounts for 1 % of the total volume of waste produced by industry and nuclear research. For the duration of the activity, the dividing line usually stands at 30 years – under 30 years, it is referred to as short-life waste and, beyond this period, long-life waste.

As a reminder, natural uranium mined from the earth's crust is composed primarily of two isotopes: 99.3 % is U-238 and 0.7 % is the fissile U-235. Fresh nuclear fuel from standard LWRs, however, has a different composition: it is made of uranium dioxide (about 95 % U-238 and 5 % U-235, after enrichment). During irradiation, most of the U-235 is fissioned, and a small fraction of the U-238 is transmuted into heavier elements (known as "transuranics").

The spent nuclear fuel from standard LWRs contains approximately:

- 3.7 % of short-lived radionuclides (majority of fission products)
  (= wastes for which partitioning and transmutation are generally not necessary)
- 0.23 % of iodine (I-129), cesium (Cs-135) and technetium (Tc-99)
  (= long-lived fission products, few of which are transmutable)
- 0.07 % of Np (neptunium), Am (americium), Cm (curium), etc
  (Np, Am and Cm = heavy minor actinides, some of which are transmutable)
- 94 % of U-238 (fertile)
- 1 % of U-235 (fissile)
- 1 % of Pu (mostly fissile Pu-239)

It is worth noting, from the above balance, that 96 % of the spent nuclear fuel is actually re-usable, if fertile material is converted into fissile. In a LWR (using U-235), 200 tons of natural U are necessary to produce 1 GWe x year, compared to 1 ton for a FBR (using U-238).

Heavy actinides are formed by neutron capture (namely: transuranic radionuclides or transuranics, also called collectively TRU). They have an atomic number greater than Uranium U*92 (that is: Np*93, Pu*94, Am*95 and Cm*96) and usually have long lives. The minor actinides (MAs) are Np, Am and Cm, whereas the major actinides are U and Pu. Uranium, if separated from the other elements, is relatively benign, and could be disposed of as low-level waste or stored for later use.

The transuranic elements have isotopes that remain highly radioactive in general for tens of thousands of years – actually they make up most of the radiotoxicity of spent fuel:

- plutonium-239 (half-life of 24 000 years): all isotopes and compounds of plutonium are toxic and radioactive (alpha emission), especially Pu-238 (86 years, highly radioactive emitter of alpha and gamma, used as neutron source or as heat / electric energy source, e.g. in space applications) – therefore, plutonium is sometimes described in media reports as "the most toxic substance known to man".
- americium-241 (430 years; highly radioactive alpha and gamma emitter): Am-241 has a relatively short half life, emitting alpha particles (helium ions) to become neptunium-237. Alpha emission from Am-241 is three times that of radium. It is also a strong gamma emitter: gram quantities of Am-241 emit intense gamma rays which creates a serious exposure problem for anyone handling the element.
- neptunium-237 (2.1 million years): Np-237 decays to form bismuth, unlike most other common heavy nuclei which decay to make lead - successive neutron captures in Pu-239 lead to Am-241, which in turn emits alpha particles to become Np-237
- curium-244 (18 years; highly radioactive neutron and alpha emitter): Cm-244 produces a large amount of neutron radiation from spontaneous fission: it is also thermogenic (it has been studied as an energy source for space applications, having a maximum energy density of 3 W/g).

As of 2000 in the EU-25, some 40 000 m^3 of radioactive waste has been generated each year. The vast majority originates from NPPs and other nuclear installations and 90 % is identified as low- and medium-level waste (LMLW). As it is, most radionuclides (notably fission products) decay rapidly, so that their collective radioactivity is reduced to less than 0.1 % of the original level 50 years after being removed from the reactor. After 400 years, the radiotoxicity of the fission products reaches the level of natural uranium, that is: for 1 ton of UO_2, a total of 5 10^5 Sv. A total of 2 million m^3 of such waste has been disposed of in the EU so far, most of it in surface or near-surface facilities. The major fuel cycle strategy for LMLW is the storage in surface or low depth for a duration of 30 - 100 years (following the local policy for plant decommissioning).

As of 2000 in the EU-25, the total volume of waste from the reprocessing of used fuel, fission products and minor actinides stored since 1950 has been about 6 000 m^3 (HLW to be vitrified). This volume is increasing at the rate of 240 m^3 a year. The disposal option for long-lived and high-level waste (including spent fuel) is deep geological disposal.

The composition of spent fuel poses specific problems that make its ultimate disposal challenging. As of today, there are three main options for the back-end of the fuel cycle:

n° 1: final disposal: once-through or open cycle, i.e. strategy of direct geological disposal of spent fuel, often with the possibility of retrievability, as is the intention in the USA and in most EU countries – notice that direct disposal requires between 3 and 4 times more space than vitrified waste disposal and that "plutonium mines" could be formed with relatively easy access after 100 years

n° 2: recycling or closed cycle, i.e. reprocessing like in France (25 years of experience), Russia and Japan as well as in China and India, also reconsidered in the USA. In a sustainable development perspective, recycling seems to be an attractive option for improving the efficiency of natural resource management and reducing radioactive waste accumulation (up to 96 % of the spent fuel material could be recovered):

- separation of U and Pu from spent fuel with no recovery of minor actinides and fission products (more than 15 % of natural resources are saved)
- vitrification in waste packages for geological disposal of high level waste (the amount of wastes is reduced by a factor 5 and the radio-toxicity by a factor 10)

n° 3: long-term storage, which consists of placing these substances temporarily in a specially laid out installation at the surface or at a low depth, pending their retrieval (it is actually an interim solution).

In the future, an increasing number of countries worldwide will be concerned about the heavy actinides and fission products that will not decay to safe levels for hundreds of thousands of years. Sustainable solutions are discussed for the long-term disposal of *high-level long-lived radioactive wastes* (in particular, spent nuclear fuel) along the following lines:

n° 4: the search for a geologic environment that will remain stable for that period (that is: reversible or irreversible disposal in deep geological formations) that goes together with the development of waste forms that can contain these elements for that period (e.g. vitrification)

n° 5: the destruction of these isotopes, that is: separation and transmutation of long-lived radioactive elements - with secular disposal (high-level and short-lived wastes) as ultimate goal. This *visionary* strategy relies on research in Generation IV (see

Sections 6 to 12) and in *Accelerator Driven Systems* (ADS) devoted to waste transmutation (following the work initiated in 1993 by Nobel Prize Carlo Rubbia).

More details about ADS are given in Appendix 3, in particular regarding the project MYRRHA (“*Multi-purpose hYbrid Research Reactor for High-tech Applications").*

GIF technology goal n° 1: Sustainability
(enhanced fuel utilisation and optimal waste management)

As far as ***fuel utilisation*** is concerned, one of the limitations of the current LWR technology might be the neutronic characteristics that reduce the capability to extract all the potential energy from the uranium fuel. This is certainly true for the “thermal neutron” designs of Generation I and II without MOX fuel, that account for the vast majority worldwide of today's nuclear fleet. In Generation I and II reactors (thermal spectrum systems), less than 1 % of the potential energy in the natural uranium fuel is used (remember: only 0.7 % of natural uranium is the fissile U-235). In the absence of reprocessing, the remainder is left unused in the spent fuel and in the uranium, depleted in U-235, that remains from the process of enriching the natural uranium in the isotope U-235.

Innovation in fuel utilisation: enhanced breeding (fast neutron spectrum systems)

Generation IV, in particular, in the fast reactor option, will harvest the above-mentioned unutilized energy. Remember that 96 % of the spent nuclear fuel is actually re-usable, if fertile material is converted into fissile. Fast reactors can convert the fertile U-238 into fissile Pu-239 at rates faster than it is consumed (breeding) making it economically feasible to utilize ores with very low uranium concentrations (see Appendix 2). As a consequence, the amount of energy extracted from the same amount of mined uranium is extended by a factor of at least 50. Even U found in the oceans could be potentially exploited (natural concentration 0,0032 mg/liter [20]). It is worth recalling that the first generation of fast reactors (built in the late 1960s) required the separation of plutonium from irradiated fuels and from reactor blankets in order to fabricate new fuels and blanket elements. The operations for the separation of various physical and chemical elements are the so-called "reprocessing" or "partitioning" operations.

As far as ***waste management*** is concerned, one of the limitations of the Generation I and II reactors might be the presence of heavy actinides and certain fission products, that are long-lived /LLFP/ and/or thermogenic, in the high-level waste from spent fuel.

- Three isotopes, which are linked through a decay process (Pu-239, Am-241 and Np-237), are the major contributors to the estimated dose for releases from the repository. They also contribute to the long-term heat generation that limits the amount of waste that can be placed in the repository. Np-237 is the most mobile actinide in the deep geological repository environment. This makes it and its two predecessors candidates of interest for destruction by nuclear transmutation. Note, however, that spent nuclear fuel is protected from theft for about one hundred years by its intense radioactivity, and therefore it is difficult to separate these isotopes without remote handling facilities.
- Certain short-lived fission products are major contributors to the repository's short-term heat load during the first decades, in particular, strontium-90 (half life of 29 years, beta emitter, emitted power = 0.93 W/g) and cesium-137 (half life of 31 years, gamma emitter, emitted power = 0.32 W/g). Their thermogenic effects, however, can be

[20] http://www.world-nuclear.org/education/phys.htm

mitigated by providing better ventilation to the repository or by providing a cooling-off period before placing them in the repository.
- Certain long-lived fission products also contribute to the estimated dose, in particular, iodine-129 (16 million years), cesium-135 (2.6 million years) and technetium-99 (210 000 years).

In fast reactors, minor actinides are produced in smaller quantities and they are more fissionable (transmutation – see Appendix 3). Full actinide management is thus possible in a fast neutron spectrum. This means much more than just the recycling of uranium and plutonium: the only high-level wastes (arising from losses in recycling/refabrication) will be relatively short-lived and in small quantities. It is worth recalling that transmutation (or burning) of plutonium is also possible in thermal spectrum LWRs (MOX or *Mix of U and Pu Oxides* or thorium fuel in MSR – see Appendix 2). The treatment of long-lived fission products, such as iodine-129, cesium-135 and technetium-99, may require a very specific approach. For example, dedicated irradiation devices in thermal or close to thermal neutron spectra may be used, since the basic resonances of these fission products are just in this energy range. The ultimate goal of innovation in this area would be to destroy all (long-lived) heavy actinides from the past and present legacy. In doing so, the ultimate obstacle from a public acceptance point of view would be removed.

Innovation in waste management: homogeneous recycling (Generation IV)

Generation IV contains several fast neutron reactor options with fully closed fuel cycles. Their strategy is called *single stratum*, that is: “infinite” recycling of Pu and minor actinides in fast reactors. The actinides are not separated (*homogeneous reprocessing*, that is: U, Pu and minor actinides or MAs together), but returned as “dirty fuel” to the reactors where they are recycled. One could say that this future recycling technology will be able to *achieve "clean waste"* (that is: relatively short-lived fission products only and in small quantities) and to *recycle "dirty fuel"* (that is: containing all transuranics and optionally some long-lived fission products) back to the nuclear power plants. This fuel cycle strategy is called *homogeneous*. It relies on fast reactors with minor actinide bearing fuel, preferably in association with an integrated on-site fuel reprocessing facility. Heavy actinides can be transmuted in fast reactors (especially for non-fissile isotopes, like Pu-240) – but also in ADS (see Appendix 3). Therefore, innovative partitioning techniques are needed that could process the major actinides (U, Pu) and the minor actinides (Np, Am, Cm) together. In addition, innovative fuel fabrication techniques will be developed ("minor actinide bearing" fuel). In doing so, the ultimate goal of Generation IV would be achieved: the new systems would produce power in the most efficient and sustainable way - maximum energy would be extracted from the ore resource (breeding) while burning the long-lived minor actinides (transmutation).

Innovation in waste management: heterogeneous recycling (dedicated systems)

However, the question remains, how to destroy all actinides from the past legacy and from the non-Generation IV reactors. This is possible following the strategy of *double strata (NPP and transmutation)*, that is: the major actinides (U and Pu) are recycled through MOX either in a PWR (Pu single recycling) or in a fast reactor (Pu infinite recycling), whereas the MAs are burnt in a subcritical / critical facility (*heterogeneous reprocessing*, that is: separating U and Pu on one side and MAs on the other side). This approach is complementary to the above homogeneous recycling strategy of Generation IV. Ambitious long-term goals are fixed with dedicated fast reactor systems: either critical (burners) or subcritical (ADS) systems. Accelerator-driven systems (ADS) are accelerator-target systems providing fast neutrons to a sub-critical reactor. This fuel cycle strategy is called *heterogeneous*. It relies on dedicated (critical or subcritical) systems to burn the MAs. Therefore, innovative partitioning techniques are needed that could separate U, Pu and the MAs into three streams: the recycled U and Pu are used in fast neutron reactors, whereas certain actinides go in dedicated fast reactors (critical or subcritical). In addition, innovative fuel fabrication techniques will be developed in order to burn the MAs in the subcritical / critical facility ("minor actinide bearing" fuel).

Note: minor actinide bearing fuels (one of the biggest challenges of Generation IV)

One of the requirements of Generation IV is the development and fabrication of advanced fuels (in particular, Pu and minor actinides bearing fuels) for fast neutron reactors and ADS, and the determination of their basic materials properties. Research focuses on both solid solution and dispersion solution fuels as well as on CERMET and CERCER fuels. The cermet fuel consists of ceramic fuel particles (usually UO_2, or, for transmutation purposes in ADS, Pu and MA without fertile uranium) in an inert metallic matrix, such as: *(Pu, Am, Cm)*O_{2-x}*.+ Mo, Mo-92, W, Cr or V*. The cercer refers to the same type of ceramic fuel but in an inert ceramic matrix, such as: *(Pu, Am, Cm)*O_{2-x}*.+ MgO*. Good candidate materials for the inert matrix phase should have a low neutron absorption cross section (in particular, a negative coolant void worth), a large thermal conductivity (preferably larger than that of UO_2), a high melting point and a good compatibility with the cladding material and the reactor coolant. In addition to these neutronics and thermo-mechanical properties, ideally, the matrix material should be easy to reprocess and/or should be a good storage medium for final disposal.

A number of large-scale irradiation campaigns are devoted to these innovative fuels, e.g.:

(1) French CEA fast spectrum reactor Phenix: SUPERFACT and FUTURIX programmes:

- SUPERFACT 1 (1986 – 1988, minor actinide MOX) - demonstration of the feasibility of minor actinides' transmutation using SFR type of oxide fuels with Am and Np:
  – two Am pins – pellets of solid solution ($U_{0.74}$, $Pu_{0.24}$, $Am_{0.02}$) O_2
  – two Np pins - pellets of solid solution ($U_{0.74}$, $Pu_{0.24}$, $Np_{0.02}$) O_2
  with the following results: transmutation ratio of circa 30 % for Np-237 and for Am-241. This irradiation provided the first results on the incineration of minor actinides in homogeneous mode. The post-irradiation examinations showed that the fuel underwent a similar microstructure evolution as standard (U, Pu)O_2 fuels. However, large helium release was measured in the Am bearing fuel (4 times more than in standard fuel) because of the production and decay of Cm-242.
- SUPERFACT 2: transmutation of long-lived radioactive products (both MA and fission products such as I-129 and Tc-99) resulting from higher burnup
- FUTURIX FTA ("*Forte Teneur en Actinides*", 2004 - 2010): irradiation in Phenix (started in May 2007) of oxide (cermet fabricated by ITU and cercer fabricated by CEA) and metallic fuels.
- FUTURIX MI ("*Matériaux Inertes*"). Testing of dispersion fuel – carbides, nitrides or refractory metals – for GFRs (e.g. U-Pu-N, U-Pu-C / Ti-N, Si-C at 900 - 1000 °C, fast neutron fluence close to 10^{27} n/m^2 and doses up to 42 dpa).

(2) European EFTTRA programme ("*Experimental Feasibility of Targets for TRAnsmutation*") in the High Flux Reactor (HFR) of Petten (NL) in the late 1990s, a collaboration of CEA, EdF, FZK, JRC-ITU, JRC-IE and NRG, focusing on cercer fuel.

(3) GIF project SFR / GACID ("*Global Actinide Cycle International Demonstration*"): aiming at demonstrating (on a significant scale) that fast neutron reactors can manage the whole actinide inventory to satisfy the Generation IV criteria (Monju fast neutron reactor).

5 OTHER DRIVERS FOR INNOVATION IN REACTOR SYSTEMS AND FUEL CYCLES: ECONOMICS, SAFETY, AND PROLIFERATION RESISTANCE (GIF TECHNOLOGY GOALS N° 2, 3 AND 4)

Prior to discussing economics, it is worth recalling what may happen to our planet *earth* according to scenarios proposed by the world climate futurologists and demography experts. The future of energy is at the heart of these discussions (Figure 4, Marchetti in 1985), especially the possible substitution of the *carbon-based economies* by the *hydrogen economy*.

To understand the issues at stake, it is essential to define what is meant by primary and secondary energy:

- *Primary energy*: may be extracted from the ground, taken from the natural environment or (by convention) be produced by a nuclear or hydraulic power plant, that is: *fossil, renewables and fissile* (used primarily for electricity/heat conversion) which are at the heart of the *Energy Policy for Europe* (EPE) [4] – see Section 2
- *Secondary energy*: is converted from primary energy into a usable form - that is: the *energy carriers.* Those are systems or substances used to transfer energy from somewhere to somewhere else or simply to generate usable kWh at one place – that is, as of today, principally: *electricity and hydrocarbons (oil products).*

One of the crucial questions today is how to reduce the greenhouse gas (GHG) emissions in the consumption of primary energies and/or how to improve their conversion into secondary energies in a sustainable way. At this point, it is interesting to recall the current world wide breakdown of primary energy by application: 18 % of the primary energy consumption goes to electricity and the rest is consumed directly as fuel (55 % for heat and 27 % for transportation). As a consequence, the solution could come in part from a more sustainable production of electricity (directly from primary energies) and/or from a new generation of energy carriers, such as synthetic fuels (originating from coal or gas, with reduced GHG emissions) and hydrogen (originating from water, without GHG emissions).

GIF technology goal n° 2: economics (minimisation of costs of MWe installed and MWth generated)

As a way of reminder, the components of the electricity production costs are:

- the investment costs (mainly related to the MWe installed) comprising
  - nuclear steam supply system (NSSS) costs
  - balance of plant (BOP) costs – "Nuclear BOP" and "Conventional BOP"
  - pre-operational costs, contingencies, decommissioning fund, interest, etc
- the operation and maintenance (O&M) costs - mainly related to the MWth generated
- the fuel cycle costs (FCC) - mainly related to the MWth generated.

As far as the nuclear steam supply system is concerned, one of the limitations of the current LWR technology (Generation I and II reactors) might be the maximum core outlet temperature (typically 285 °C for BWR and 320 °C for PWR) and the size of the reactor pressure vessel, dictated, amongst others, by the power density that lies between 30 and 50 $MWth/m^3$ (PWR and BWR, resp). The relatively low core outlet temperature means that the thermal efficiency is limited to approximately 35 %. Another limitation of the current LWR technology is the fact that only electricity (no process heat) is generated.

As far as *economics* is concerned, the limitations of the current LWR technology have been driving a number of industrial initiatives worldwide. The following challenges have been identified in the *European Utility Requirement* [21] (EUR) association and in the "*Nuclear Power 2010*" programme [22] (US-DOE driven *International Near-Term Deployment* Group):

- standardised designs to speed up licensing, reduce capital cost, construction time
- more robust designs, making the plants easier to operate and maintain
- use modern man-machine interfaces (digital I&C, ergonomic control rooms, CAD)
- longer operating life (typically 60 years) and minimal effect on the environment
- capacity factor of 90 %, which is much more than all other means of electricity
- enhanced safety and reliability (in particular, reduced probability of severe accidents).

Innovation in economics: plant management and higher thermodynamic efficiency

Generation IV aims at the reduction of all components of the electricity production costs. As far as the investments are concerned (that is: costs of MWe installed), this means:

- constructability (less than 48 months); modular plants (additional modules can be added), life time of at least 60 years; capacity factor of at least 92 %; load following
- high-performance turbines for generation of electricity, using high-temperature Brayton cycles with He or supercritical CO_2 (rather than Rankine steam cycles)
- digital instrumentation and control /I&C/, easier ISI, maintenance and repair
- economics of non-electricity energy products; distributed generation due to small size
- simpler infrastructures for materials and fuel fabrication; development of high burn-up fuel (up to 200 GWd/mtHM in VHTR); waste (minor actinide) management.

As far as the costs of the MWth generated are concerned, Generation IV aims at higher thermodynamic efficiencies. Using other coolants, higher temperatures can be achieved, such as 400 – 600 °C for CO_2, 500 – 700 °C for liquid metals and 700 – 900 °C for helium gas (target of 1000 °C). For example, a core outlet temperature of 900 °C can give a thermodynamic efficiency higher than 45 %. Large power densities up to 300 MWth per m3 allow compact designs, which is beneficial for the NSSS cost. Another important innovation is the cogeneration of heat and power (electricity), which enables nuclear fission to penetrate into the market of heat and transport (55 % and 27 % of primary energy consumption worldwide – see above). The general aim is not only to produce electricity in a more efficient and sustainable way, but also to develop other applications such as: the supply of high-temperature heat for the (petrochemical) industry and the production of a new generation of energy carriers with no GHG emissions (e.g. synthetic fuel and hydrogen from water). It should also be recalled that the overall economics is closely related to safety and reliability.

Should the world adopt nuclear fission systems to produce process heat (in addition to electricity), the growth rate of nuclear energy would greatly accelerate. With an accelerated growth for nuclear energy, the uranium resources would be depleted in a few decades, with the once-through fuel cycle currently in use in most countries. Therefore, the fast neutron reactors (with their recycling capabilities) are at the top of the GIF agenda.

The Working Group of GIF on "*Economics Modelling*" (EMWG) [23] provides useful guidelines.

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[21] http://www.europeanutilityrequirements.org/

[22] http://www.ne.doe.gov/np210/neNP2010a.html

[23] www.gen-4.org/Technology/horizontal/EMWG_Guidelines.pdf

GIF technology goal n° 3: safety and reliability (robust safety architecture and enhanced EUR requirements)

As far as reactor ***safety*** is concerned, one of the possible limitations of the current nuclear fission technology (Generation I and II reactors) might lie in the last defence against hypothetical severe accidents. As a reminder, nuclear safety is defined by the IAEA as "*the achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in the protection of workers, the public and the environment from undue radiation hazards*". A particular aspect of reactor safety is the need to guarantee the safe shut-down of the fission chain reaction. About 7 % of heat output is typically derived from delayed neutron fission, the decay of fission products and other transuranic elements. When the chain reaction is stopped, this source of heat, known as decay heat, remains. This heat (about 200 MWth of a 3000 MWth plant) has to be adequately removed after the shut-down of the reactor. This is usually done through "*active*" systems.

It should be recalled that from a safety point of view, fast neutron reactor cores have the fundamental characteristic of not being in the most reactive situation. That is: there is a risk of power excursion in accident situations if the void neutronics reactivity coefficient is positive and cannot be counter-balanced by negative Doppler and other feedback mechanisms. Here also, active systems are put in place. These systems rely on the active electrical and mechanical control of equipment such as sensors, valves, pumps, accumulators, heat exchangers, and backup power supplies.

It is also recognised that the multiple parallel redundancies that are built into the current designs add to the complexity of the systems and to the construction and maintenance costs. A holistic approach, covering the entire lifecycle of the reactor system, could help simplify and improve the safety architecture.

Innovation in safety: a "robust" architecture ("dealt with" and "excluded" core disruptive accident initiators)

Generation IV opts for a "robust" safety architecture (that is: free of dominant vulnerabilities), integrating all four GIF technology goals in the design ("built-in" features, not "added-on"), for example, through the following features:

- selection of coolants that are chemically inert and of materials that are resistant to high-temperatures and pressures under high irradiation conditions (strong emphasis on quality assurance throughout the entire design process) – see Figure 5
- tough design requirements, such as total negative reactivity coefficients (Doppler, temperature, void, etc) for all accidental transient conditions (ULOF, ULOHS, UTOP)
- implementation of defence-in-depth including the fourth level (management of core disruptive accidents / CDA) and coherent treatment of internal and external events
- take advantage of new safety assessment instruments (e.g. "risk-informed" approach as a complement to the deterministic approach; Objective Provision Trees /OPT/ and Lines of Protection /LOP/).

Generation IV will provide a "robust" safety demonstration, in particular, regarding "dealt with" and "excluded" postulated single initiating events or CDA initiators

(i) "dealt with" CDA initiators (range of different scenarios for design basis conditions /DBC/ and design extension conditions /DEC/)

- demonstration of defence-in-depth strategy in a manner that is exhaustive, progressive, fault tolerant (no cliff-edge effects), forgiving (e.g. satisfactory grace period) and well balanced – in VHTR, the coated particle provides excellent containment (first barrier)
- mitigation of consequences of certain severe accidents (DEC): e.g. air or coolant ingress; low pressure core melt accident (to be managed either by in-vessel melt retention or by ex-vessel melt cooling, e.g. through spreading on a core catcher)
- reduced reliance on human actions to mitigate off-normal conditions, e.g. use of passive decay heat removal systems such as core catchers, thereby providing sufficiently large grace period for implementing corrective actions

(ii) "excluded" CDA initiators, beyond the accidents considered in the DECs (physically impossible or to be "practically eliminated" by design) – *Residual Risk* /RR/

- “practical elimination” of a limited number of initiators, sequences or phenomena associated with the extremely low RR (e.g. leading to unacceptable core damage and/or uncontrolled large early releases of fission products)
- demonstrate that sufficient provisions are foreseen to make these events (e.g. where the consequences cannot be realistically managed) practically impossible to happen
- examples of initiators to be "practically eliminated": failure of the core support structure, excessive core compaction, transportation of large bubbles in primary coolant system of Sodium Fast Reactors.

As far as ***reliability*** is concerned, possible limitations of the current nuclear fission technology are related to capacity factor, man – technology – organisation (MTO) interface, radiological impact on workers and population (radiation dose limits, ALARP principle /*as low as reasonably practicable*/), in-service inspection, etc, as it is discussed in the EUR specifications. The greatest departure from Generation II designs will be that many Generation IV systems incorporate passive or inherent safety features which require no active controls or operational intervention to avoid accidents in the event of malfunction. Such passive systems are simple systems that rely on gravity, natural convection, spring or compressed gas supported systems, or resistance to high-temperatures. The assessment of these systems, however, is difficult because of the limited feedback of experience. Passive safety, of course, is not an objective *per se*: it should be efficient, reliable and economic.

Innovation in reliability: enhanced EUR requirements

The reliability of Generation IV reactors during plant life will also be improved through the development of appropriate fuels, materials, components and systems for primary and secondary circuit, that is:

- very tough design and operation requirements that go beyond the standard practices, thereby optimising the thermal, neutronic, mechanical and chemical balance
- very tough fuel and materials requirements that could go further than the European Utility Requirements (EUR), such as operating lifetime of 60 years, capacity factor of 90 % (= 8000 hours operation per year), instrumentation and control, human factor
- in-service inspection /ISI/ (e.g. RPV integrity, surveillance programme, non destructive testing /NDT/ systems).

GIF technology goal n° 4: proliferation resistance and physical protection (impractical separation of plutonium)

As far as proliferation resistance /PR/ and physical protection /PP/ (also called together nuclear <u>security</u>) are concerned, one of the limitations of the current nuclear fuel cycles

(Generation I and II reactors) might be the risk of uncontrolled U enrichment and Pu extraction (the so-called "sensitive" technologies). As a reminder, *Proliferation Resistance* is defined by the IAEA as that "*characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States intent on acquiring nuclear weapons or other nuclear explosive devices*". *Physical Protection* is defined as that "*characteristic of a nuclear energy system that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices, and the sabotage of facilities and transportation, by sub-national entities and other non-host State adversaries*".

Remember that pure Pu-239 (from short term n-irradiation of U-238) can be used as weapon material. The critical mass for an unreflected sphere of plutonium is 16 kg: the critical mass of plutonium is only a third of that of uranium-235. In addition to plutonium, the fissile isotopes of americium and neptunium are also potentially usable in weapons and, therefore, raise proliferation concerns. For example, the critical mass for an unreflected sphere of Am-241 is approximately 60 kg. Special care is also needed during the transportation of Pu and/or fresh MOX from the reprocessing plants to the NPP.

Significant risks are also related to the nuclear stockpiles that are at the heart of the discussions about Non-Proliferation. Large stockpiles of "weapons-grade" plutonium were built up by both the Soviet Union and the USA during the Cold War. The US reactors at Hanford and the Savannah River Site in South Carolina produced more than 100 tons. It is estimated that another 170 tons of military plutonium have been produced in Russia, with 300 tons accumulated worldwide. Since the end of the Cold War, these stockpiles have become a focus of nuclear proliferation concerns. In 1997, both the USA and the Russian Federation recognized the reality that their nuclear stockpiles vastly exceeded their post-Cold-War military requirements. As a consequence they pledged that hundreds of tons of their fissile material stockpiles will never again be used in nuclear weapons. In 2000, both the USA and Russia concluded a surplus plutonium disposition agreement: each country would dispose of 34 tonnes of weapon-grade plutonium by the year 2020 (for example, by mixing it with uranium to form MOX fuel to be used in power plants of Generation III or IV type).

Innovation: impractical separation of plutonium

Generation IV, especially with the option of closed fuel cycle with on-site reprocessing, will make illicit diversion from within the process highly impractical. Two characteristics are of particular interest:

- contamination of MAs increases fuel activity and makes Pu less attractive for weapons
- high concentrations of Pu-238 (after long n-irradiation) make Pu useless for weapons.

The fast neutron reactors of Generation IV and the P&T systems will burn some of the excess Pu and other fissionable materials. It is clear, however, that this burning will not eliminate the need neither for the disposal of ultimate radwaste in deep geological formations, nor for an international non-proliferation regime, but they make the task easier by segregating and consuming the actinides as they are created. Finally, innovative fuel fabrication techniques will make it difficult to extract fissile material from spent fuel (but this may complicate material accounting in the international safeguards system).

The Working Group of GIF WG on "*Proliferation Resistance and Physical Protection* (PR&PPWG) [24] provides useful guidance.

[24] http://nuclear.energy.gov/pdfFiles/PRPP.pdf

6 GENERATION IV: A TECHNOLOGY BREAKTHROUGH / FROM 2000 (START OF RESEARCH) TO 2030 (START OF DEPLOYMENT) / "VISIONARY" INNOVATION

Building upon Generation III but with a longer term perspective, a US DOE driven High Level Policy Group met in January 2000 to launch the *Generation IV International Forum* [25] (GIF). A task force was organised with the aim of selecting a number of innovative nuclear reactor technologies or *systems* that could be deployed by 2030.

The GIF Charter was signed in July 2001. The GIF members were originally a group of nine countries: Argentina, Brazil, Canada, France, Japan, South Africa, South Korea, the UK, and the USA. Switzerland signed in February 2002, and the European Atomic Energy Community (Euratom) in July 2003. They are governments, usually represented by national nuclear research organisations. The Russian Federation and the People's Republic of China signed the GIF Charter in November 2006, bringing the number of members of the organization to 13. A *Senior Industrial Advisory Panel* (SIAP) has been nominated to discuss the industrial deployment aspects. The *Technical Secretariat of the GIF* is with the OECD/NEA [26], Paris: they also support the work of the system "cross-cutting" committees.

The legal and administrative structure of GIF [27] consists of

- one *Framework Agreement* at the intergovernmental level, signed by 9 Members (Canada, France, Japan, South Korea and the USA in February 2005; Switzerland; Euratom in May 2006; China in January 2008 and South Africa in April 2008)
- 6 *System Arrangements* (SFR signed by a first group in February 2006 and later in February 2007; VHTR, GFR and SCWR signed by a group in November 2006)
- several *Project Agreements* (PA) per system (as of May 2008, three PAs have been signed for SFR / "*Advanced Fuel*", "*Component Design and BOP*" and "*GACID*"/ and two PAs signed for VHTR /"*Fuel and Fuel Cycle*" and "*Hydrogen Production*"/).

A total of six Generation IV systems were selected in July 2002, following the evaluation of more than 100 different nuclear energy concepts by more than 100 experts from a dozen countries. As a result of the Generation IV Roadmap exercise [28] (started in March 2001 and terminated in December 2002), a "*Technological Roadmap for Generation IV Nuclear Energy Systems*" was proposed. The six systems and the respective System Research Plans (SRP) are described in this Roadmap. The total budget of the GIF partners needed for the six systems, before a first prototype can be built, is estimated at circa US $ 1 billion.

Of particular interest are the four "*Technology Goals for industry and society*" of Generation IV (mentioned in Sections 4 and 5, and elaborated on in the Table below, *"Translation of…"*), that is: the criteria for inclusion of a reactor design for consideration:

- sustainability: e.g. enhanced fuel utilisation and optimal waste management
- economics: e.g. minimisation of costs of MWe installed and MWth generated
- safety and reliability : e.g. robust safety architecture and enhanced EUR requirements
- proliferation resistance and physical protection: e.g. impractical separation of Pu.

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[25] http://nuclear.energy.gov/genIV/neGenIV1.html

[26] http://www.g4if.org/gif/

[27] http://www.gen-4.org/GIF/Governance/index.htm

[28] http://www.gen-4.org/Technology/roadmap.htm

Translation of the 4 Technology Goals in industrial solutions and research challenges

GIF technology goals	***Technological implementation***	**Industrial solution**	**Research challenges**
Goal 1 Sustainability	*Enhanced fuel utilisation*	fast neutron reactors (e.g. SFR, GFR, LFR)	burn maximum of the spent fuel (out of 96 % that is re-usable)
	Optimal radioactive waste management (full actinide management)	"clean waste" (and "dirty fuel") (breeder versus burner)	➢ single stratum (homogeneous) ➢ double strata (heterogeneous)
Goal 2 Economics	*Cost of MWe installed (level of financial risk comparable to other energies)*	➢ constructability (< 48 months) ➢ modularity (turnkey plants)	➢ power plant simplification ➢ integrated infrastructures for nuclear fuel cycle
	Cost of MWth generated (clear life-cycle cost advantage (externalities)	➢ cogeneration of heat & power ➢ lifetime 60 years and capacity factor 92 %	➢ high core outlet T ➢ cogeneration (high T process heat) for generation of syn-fuels and hydrogen
Goal 3 Safety	➢ *robust safety architecture (very low likelihood & degree of reactor core damage)*	➢ ALARP ➢ elimination of the (technical) need for off-site emergency response	"practical elimination" of certain severe accident sequences ("dealt with" versus "excluded" CDAs)
and Reliability	➢ *enhanced EUR ("European Utility Requirements")*	optimum thermal, neutronic, mechanical and chemical balance	very tough fuel and materials requirements
Goal 4 Proliferation Resistance (PR)	➢ *very unattractive route for diversion of fissionable materials*	"dirty fuel" (and "clean waste")	make separation of Pu impractical (e.g. homog. reprocessing)
and Physical Protection (PP)	➢ *physical protection against acts of terrorism*	set up security barriers, similar to safety in DiD	➢ evaluation of PP and PR threats ➢ integral design

The GIF systems selected are as follows: three fast spectrum systems (SFR, GFR, LFR), two thermal spectrum systems (VHTR, MSR) and one thermal / fast system (SCWR). Details are given about the specific research plans [29]and the major challenges for each system in Sections 7 to 12. Comparative Tables are given in Figures 14, 15 and 16 for all six Generation IV systems (Figure 15 contains also data for the PWR and the Fusion reactor), focussing on:

- industrial applications (cogeneration of heat & power, plus full actinide management)
- environmental challenges for fuels and structures (in- and outlet core temperatures, neutron exposure, system's pressure)
- selection of material classes for fuels and structures.

[29] http://www.gen-4.org/PDFs/annual_report2007.pdf

Here is the list of the six GEN IV systems, with some of their main characteristics:

Very high-temperature gas reactors (VHTR)
- cogeneration of high-temperature process heat and electricity (efficiency 45 – 50 %, core outlet temperature of 1000 °C); no actinide management - once through cycle / reference power = 600 MWth / 300 MWe / earliest delivery 2020

Sodium cooled fast reactors (SFR)
- electricity production and full actinide management (enhanced fuel utilisation, efficiency close to 40 %, core outlet temperature of 550 °C) / reference power = modules of 50 – 150 MWe or plants of 600 - 1500 MWe / earliest delivery 2020

Gas-cooled fast reactors (GFR)
- cogeneration of electricity and process heat (enhanced fuel utilisation, efficiency close to 45 %, core outlet temperature of 850 °C); full actinide management / reference power = 1000 MWe / earliest delivery 2025

Supercritical water cooled reactors (SCWR)
- electricity production at high-temperatures (improved economics, next step in LWR development, efficiency close to 45 %, core outlet temperature of 510 °C); full actinide management in the fast version / reference power = 1700 MWe / earliest delivery 2025

Lead-cooled fast reactors (LFR)
- cogeneration of process heat and electricity (full actinide management, a possible small turnkey plant, efficiency close to 45 %, core outlet temperature of 800 °C) / reference power = batteries of 10 – 100 MWe and plants of 300 - 600 MWe / earliest delivery 2025

Molten salt reactors (MSR)
- cogeneration of process heat and electricity (full actinide management, efficiency close to 45 %, core outlet temperature of 800 °C, burning of TRU, breeding in thermal spectrum using Th and in fast using U-Pu) / reference power = 1000 MWe / earliest delivery 2025.

Equally important is the individual planning for each GIF system, that consists of three phases, extending over a total period of 12 – 20 years, starting from 2003, namely:

- the *viability* phase (basic concepts, technologies and processes are proven out under relevant conditions, with all potential show-stoppers identified and resolved - pre-conceptual design) - between 5 and 15 years needed
- the *performance* phase (engineering scale processes, phenomena, and materials capabilities are verified and optimized under prototypical conditions - conceptual design) - between 5 and 10 years needed
- the *demonstration* phase (preliminary design – in view of attracting commercial deployment) - between 3 and 6 years needed.

It should be noted that only phases 1 and 2 (*viability* and *performance*) are covered by the GIF collaboration agreements. The execution the phase 3 (*demonstration),* which could contain commercial aspects, is left to the individual GIF members. To understand the GIF planning, it is first necessary to define what the various (usually 6) design phases are, following the sequence that is applied to most large-scale industrial projects, that is:

- Pre-conceptual design (*viability phase)*
- Conceptual design (*performance phase)*
- Preliminary design (*demonstration phase*)
- Engineering design
- Final project
- Construction.

The first two phases correspond, in fact, to the research and development part of the RD & DD cycle, whereas the last four phases correspond to the demonstration and deployment part.

Safety considerations are naturally very important in the GIF programme, as is seen, for example, in the list of milestones or "endpoints" proposed in their roadmap for the *Viability* and *Performance* phases (particularly in the milestones n° 4 to 9, *listed in italic below*):

(a) Viability Phase:
1. Pre-conceptual design of the entire system, with nominal interface requirements between subsystems and established pathways for disposal of all waste streams
2. Basic fuel cycle and energy conversion (if applicable) process flow-sheets established through testing at appropriate scale
3. Cost analysis based on pre-conceptual design
4. *Simplified PRA for the system*
5. *Definition of analytical tools*
6. *Pre-conceptual design and analysis of safety features*
7. *Simplified preliminary environmental impact statement for the system*
8. *Preliminary safeguards and physical protection strategy*
9. *Consultation(s) with regulatory agency on safety approach and framework issues.*

(b) Performance Phase:
1. Conceptual design, sufficient for procurement specifications for construction of a demonstration plant, and with validated acceptability of disposal of all waste streams
2. Processes validated at scale sufficient for demonstration plant
3. Detailed cost evaluation for the system
4. *PRA for the system*
5. *Validation of analytical tools*
6. *Demonstration of safety features through testing, analysis, or relevant experience*
7. *Environmental impact statement for the system*
8. *Safeguards and physical protection strategy for system, including cost estimate for extrinsic features*
9. *Pre-application meeting(s) with regulatory agency.*

As a complement to the GIF, which is rather "systems supplier" oriented, another important international initiative was launched, also in 2000, by the IAEA, with a more "end-user" oriented prospective. This is the "*INternational PROject on innovative nuclear reactors and fuel cycles*" (INPRO). It was proposed by the President of the Millennium Summit and confirmed by the UN General Assembly of 2001 [30]. As of December 2007, INPRO has 28 members or entities: Argentina, Brazil, Bulgaria, Canada, Chile, China, Czech Republic, France, Germany, India, Indonesia, Republic of South Korea, Netherlands, Pakistan, Russian Federation, South Africa, Spain, Switzerland, Turkey + European Commission (Euratom) +

[30] http://www.iaea.org/OurWork/ST/NE/NENP/NPTDS/Projects/INPRO/index.html

Armenia, Morocco and Ukraine in 2004 + United States, Slovakia and Japan + Belarus and Kazakhstan. Taking account of the Agency's unique mandate in the field of nuclear technology, safety and safeguards, the IAEA General Conference has invited all interested Member States to jointly consider innovative actions in nuclear reactors and fuel cycles that use economically competitive technology that minimises the risk of proliferation and the impact on the environment (time horizon = 2050).

INPRO takes the long-term view that nuclear energy should be considered in the broader prospective of future energy needs, and addresses the problems from the point of view of potential users (especially in developing countries, for example, with small grids and/or desalination needs) by identifying their specific needs in INS ("*Innovative Nuclear Systems*"). INPRO does not address any specific technology. They are producing comprehensive catalogues of basic principles for assessment methodologies in 6 areas (*economics, environment, safety, waste management, proliferation resistance and RTD infrastructures*). They also publish "*User Requirements*", reflecting, in particular, the specific needs of developing countries interested in nuclear cogeneration of heat and power. Two IAEA documents are of particular interest for countries planning the first NPP: "*Basic infrastructure for a nuclear power project*" (TECDOC 1513, June 2006) and "*Potential for sharing nuclear power infrastructure between countries*" (TECDOC 1522, October 2006).

Integration of all four Technology Goals in the design of Generation IV systems

As a reminder, when a material is irradiated, all of its physical properties can change. Physical dimensions as well as the electrical, mechanical, magnetic, thermo-physical and other properties can each change. In high-dose irradiation, each atom may be displaced from its lattice site many times, the standard measure of which is the displacements per atom (dpa). One dpa corresponds to approximately 180 days of irradiation in a thermal spectrum reactor and 12 days in a fast spectrum reactor.

The new and more demanding operating conditions foreseen in innovative reactor systems arise from

- fuel cycle innovations: for example, fast spectrum neutron fluxes could result in radiation dosages up to 150 dpa on cladding materials compared to about 15 dpa for thermal spectrum
- the need of robust structural materials (cladding, in-core and out-of-core materials)
- the need for higher outlet temperatures for greater thermal efficiency (from about 285 - 320 °C in LWRs to about 510 °C in SCWRs and 1000 °C in VHTRs)
- novel coolant possibilities, such as supercritical water, liquid sodium or gaseous helium, which interact more stressfully with materials comprising both the reactor core and the balance of plant than conventional light or heavy water coolants.

For each GIF system, a number of research actions are carried out by the GIF members in their national and international programmes (in particular, Euratom) – see Table above and Sections 7 to 12. One of the biggest challenges of GIF, however, is the integration of all four Technology Goals in the systems design ("built-in" features, not "added-on"). To achieve the Technology Goals, it is important to identify the key parameters and to assess their role in both the pre-conceptual and the conceptual design phases.

For example, the specification range for the *power density* is a key parameter. This value affects:

- sustainability (e.g. fuel cycle with sufficient dynamics and minimising the fuel needs for long-term deployment)
- economics (e.g. minimisation of fuel inventory, of Am production, of fuel cycle cost, compactness of the primary vessel)
- safety (e.g. decay heat removal in the case of a depressurisation event).

Economics and sustainability call for higher power densities and safety for lower values. Indeed, cores should be designed to achieve a low pressure drop value in order to facilitate coolant (gas or liquid metal) circulation in any situation. The tentative range between 50 and 100 $MWth/m^3$ appears to be a good compromise, in between PBMR values of about 7 $MWth/m^3$ and classical LMFBR ones (> 200 $MWth/m^3$). For example, for GFR, a 100 $MWth/m^3$ power density has been selected in order to achieve a reasonable in-core plutonium inventory and also as an acceptable compromise between economics and safety considerations.

Another key parameter in all innovative reactor systems is the *fuel cycle*, that is:

- fabrication: the development of an innovative fuel with a high fissile-atom density, able to sustain high levels of operating temperatures, fast spectrum, high burn-up, and with a high capacity to confine fission products at high-temperature (> 1600 °C)
- back-end: the development of the associated fuel cycle technologies with the aim to implement the integral management of actinides (transmutation).

The behaviour of the *reactor materials* is yet another key parameter. They will be subject to higher levels of hydrostatic, thermo-chemical, and radiolytic stresses over a longer period than materials in currently operating reactors. Remember that the expected lifetime of the future reactors should be 60 years. Many of these materials issues are shared by projects whose primary goal is the production of innovative energy carriers (e.g. hydrogen from high-temperature thermo- or electro-chemistry) and, interestingly, the nuclear fusion reactor development, including the *International Thermonuclear Experimental Reactor* (ITER).

Research in this domain focuses increasingly on the interaction between numerical modeling and experimental campaigns. Multi-scale multi-physics modeling is a powerful support to predicting the behaviour of fuels and materials under extreme conditions (Figure 6).

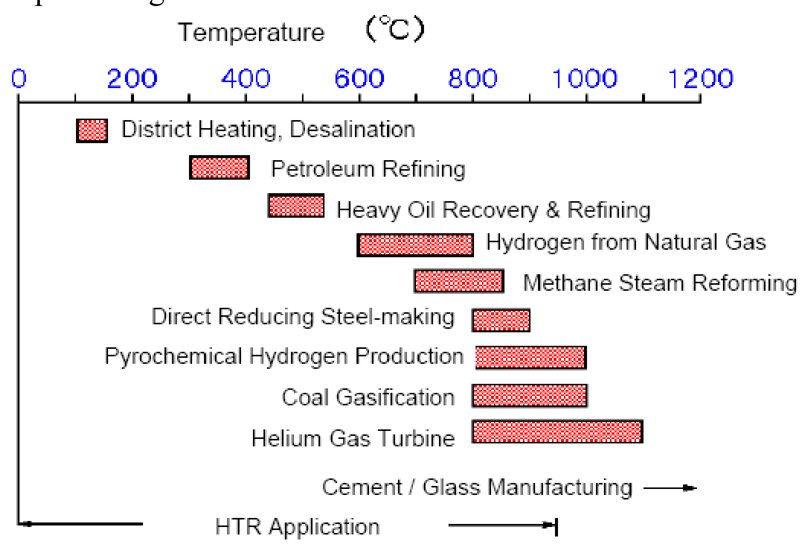

Figure 3 – Temperature range of required heat for various industries, in particular, hydrogen production / nuclear CHP (cogeneration of heat and electricity) below and above 850°C

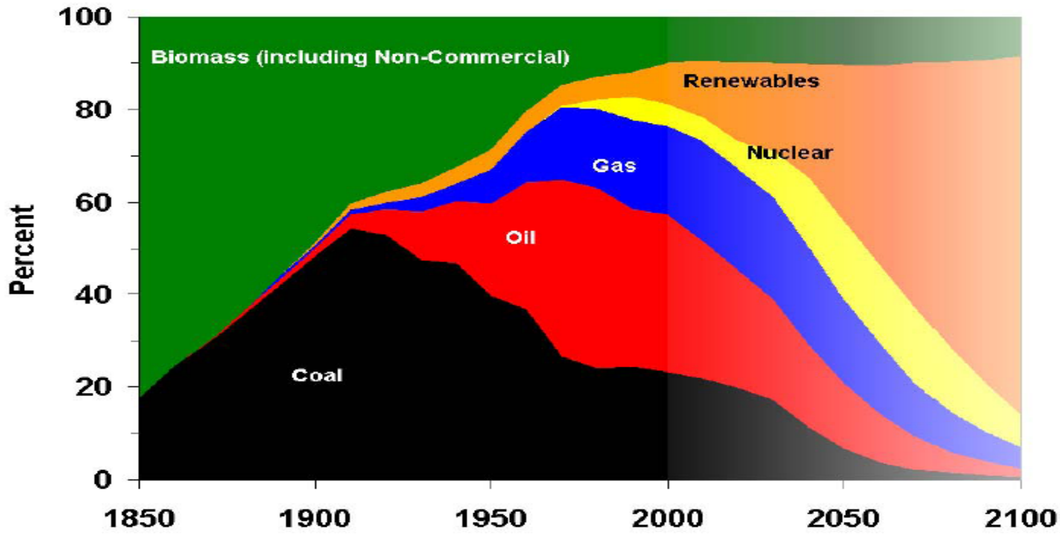

Figure 4 – Evolution of global primary energy / substitution of fossil, fissile, renewables (a great variety of energy scenarios have been developed since C. Marchetti, IIASA,1985)

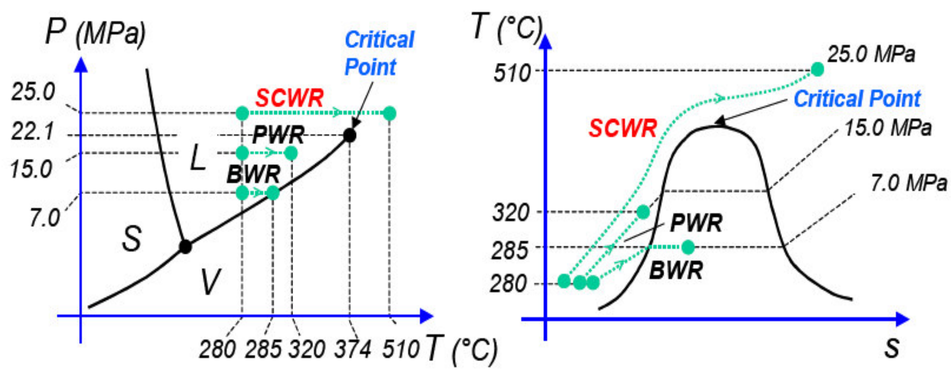

Figure 5 - Challenges for materials under extreme pressures, temperatures and irradiation: embrittlement, swelling, creep, radiolysis, stress corrosion cracking (IGSCC, IASCC)

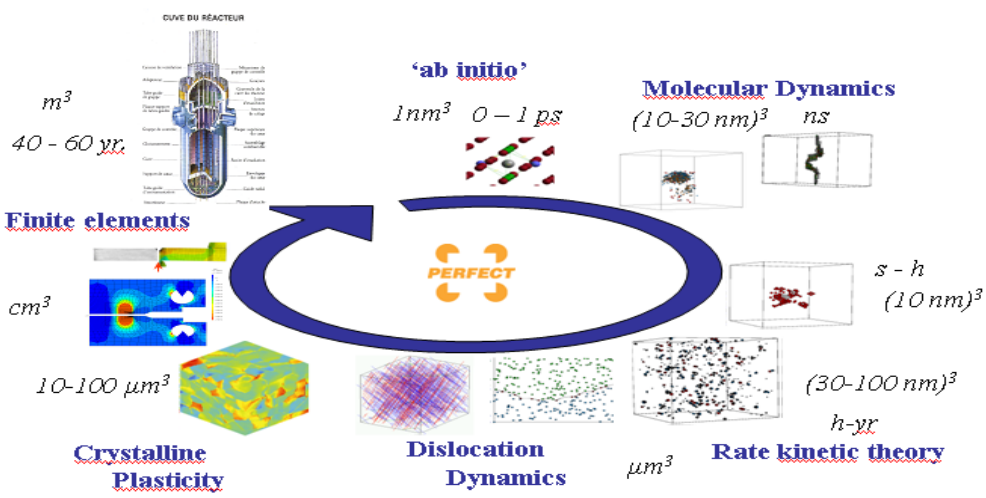

Figure 6 - Multi-scale (ps => yr // nm => m) multi-physics numerical tools for irradiation damage prediction, as a support to scaling up from laboratory to reactor conditions

7 VERY HIGH-TEMPERATURE GAS REACTORS (VHTR) / COGENERATION OF HIGH-TEMPERATURE HEAT AND ELECTRICITY (EFFICIENCY 45 - 50 %)

The very high-temperature gas reactor (VHTR) system is a thermal spectrum neutron flux, graphite-moderated, helium-gas-cooled reactor with an outlet temperature of 900 °C or higher (goal of 1000 °C) – Figure 7. The VHTR system features a once-through open fuel cycle for the uranium. As defined in the Generation IV Technology Roadmap, the missions of the VHTR are primarily the cogeneration of electricity and hydrogen, as well as other high-temperature process heat applications (e.g. petrochemical processes, synthetic fuels from coal and gas). In particular, the VHTR core outlet temperature is sufficiently high to produce hydrogen from only heat and water by using thermo-chemical, electro-chemical (electrolysis) or hybrid processes, as well as from heat, water and natural gas by steam reforming.

The potential for high fuel burn-up (150-200 GWD/mtHM), large thermal inertia, inherent passive safety features (for both the fuel and the reactor), low operation and maintenance costs, as well as modular construction, constitute major advantages in terms of commercial deployment. Owing to the significant past experience accumulated with HTRs in several countries, the system is expected to be available for commercial deployment by 2020.

VHTR specific research plans have been arranged into 6 Project Arrangements:

1. System Integration and Assessment (SI&A)
2. Computational methods validation and benchmarks (CMVB)
3. Fuel and fuel cycle (FFC)
4. Materials (MAT) and components
5. Hydrogen production technologies and nuclear process heat applications (HP)
6. Helium Turbine (High-performance turbo-machinery) and BOP.

The basic technology for the VHTR has been established in former high-temperature gas reactors of the (advanced) Generation I type such as: the German AVR and THTR prototypes as well as the US Peach Bottom and Fort Saint Vrain plants (Appendix 2).

MAJOR CHALLENGES OF THE 6 R&D PROJECTS RELATED TO VHTR

1. System Integration and Assessment (SI&A)
The VHTR is the next step in the evolutionary development of high-temperature reactors. The reference reactor thermal power is set at a level which allows for passive safety, utilizing inherent safety features, estimated to be up to 600 MWth.

Two reference concepts, both aiming at a reactor lifetime of 60 years, are available for the electric power conversion of the VHTR:

- *indirect cycle* (process heat version), such as the French ANTARES (600 MWth / 280 MWe) – as well as GFR (see Section 9): primary circuit of helium (T inlet of 400 and T outlet of 850 °C for HTR and up to 1000 °C for VHTR, under a system's pressure of 7 and 6 MPa, resp.), compact IHX design with nominal heat load of 600 MWth, secondary circuit with a gas mixture Brayton cycle turbine (80 % N_2 + 20 % He), steam generator, tertiary (closed) circuit with a steam cycle turbine to produce 300 MWe – efficiency = 45 - 50 % / the US NGNP has similar characteristics
- *direct cycle* such as PBMR (270 MWth / 115 MWe) and the original US/Russian GT-MHR (600 MWth / 285 MWe), with a direct Brayton cycle (direct flow of heated

helium from the reactor vessel): helium gas turbine, electric generator, and gas compressor located on a common long vertically oriented shaft supported by magnetic bearings – efficiency = 45 – 50 %.

As far as the core configuration is concerned, there are two reference concepts for the VHTR, both with a power density of approximately 7 $MWth/m^3$:

- *prismatic blocks* such as the current Japanese HTTR and new concepts under development, such the future French ANTARES and US GT-MHR
- *pebble bed* such as the past European AVR, the current Chinese HTR-10 and the future South-African project PBMR.

2. Computational methods validation and benchmarks (CMVB)

Computational methods development and validation in the areas of thermal-hydraulics, thermal mechanics, core physics, and chemical transport are major activities for the assessment of the reactor performance in the two core configurations (prismatic and pebble bed) under all conditions (normal, incidental and accidental). Code validation will be assessed through benchmark tests and code-to-code comparison, from basic phenomena to integrated experiments, supported, in particular, by HTTR tests or by past operational reactor data (e.g. AVR, Fort St Vrain, etc.). Normal and abnormal operating analyses will be performed, including criticality safety, coupled neutronics and thermal hydraulics, flow mixing in the hot plenum, air ingress, fission product transport and plate-out, pressurized or depressurized loss of coolant accident, and seismic behavior.

3. Fuel and fuel cycle (FFC)

The *fuel* consists of highly corrosion-resistant TRISO microspheres with a diameter of less than 1 mm, containing UO_2 (standard option), or uranium oxycarbide (UCO) or thorium oxide (ThO_2). TRISO, meaning tri-isotropic, refers to a multi-layered *cladding* or coating with four layers of three isotropic materials. The four layers are a porous buffer layer made of carbon, followed by a dense inner layer of pyrolytic carbon (PyC), followed by a ceramic layer of SiC (or advanced refractory ZrC) to retain fission products at elevated temperatures, followed by a dense outer layer of PyC. It is worth recalling that silicon improves the creep resistance above 600 °C and thus gives the TRISO particle more structural integrity.

The TRISO microspheres or coated particles need to be qualified for relevant service conditions (that is: 1300 °C). R&D will increase the understanding of standard design UO_2 kernel and examine the use of UCO kernels and ZrC coatings for enhanced burn-up capability (up to 30 % FIMA), reduced fission product permeation and increased resistance to core heat-up accidents (above 1600°C).

There is some flexibility in fuels, but no recycling, at least not in the initial phase. The burn-up target is in the following range: FIMA 15 to 20 %, i.e. 4 to 5 times more than in a standard reactor – FIMA means *Fission per Initial Metal Atom* (remember: 1 % FIMA is circa 10 GWd/mtHM). The fuel cycle will initially be a once-through very-high burn-up cycle utilizing low enriched uranium (LEU, < 20% U-235). The potential operation with a closed fuel cycle will be assessed, as well as potential symbiotic fuel cycles with other types of reactors (especially LWRs) for waste reduction. In the longer term, the incineration of Pu or/and MA is under consideration as oxides, with or without dilution in another matrix. As a consequence, R&D will examine spent fuel treatment and disposal, including used graphite management, as well as the deep-burn of Pu and MAs in support of a closed fuel cycle.

4. Materials and components (project shared with GFR)

It is worth recalling that most of the reactor core structural elements are made from graphite to provide neutron moderation as well as thermal and neutron shielding. The choice of the graphite grade is thus very important because of its impact on the dimensional stability of the core structures. The main challenge for the graphite materials is the irradiation temperature which rises up to 950 °C, which is far beyond the conventional ones (550 °C).

The main challenge for the structural metallic materials of the primary circuit is the corrosion in an impure helium environment. The nature and concentration of gaseous impurities in helium are dependent upon the rate of pollution (e.g. air or water ingress), the release of corrosion products from materials of the primary circuit and the efficiency of the purification process.

Materials - The RPV material currently considered for the "non-cooled (hot)" vessel concept (such as adopted for the GT-MHR reactor) is Cr-Mo steel with modified 9Cr1Mo steel grades similar to ASME Grade 91. This RPV is intended to operate at 400 – 460°C with potential temperature excursions up to 570°C and a lifetime of 60 years. The expected dose is 7 – 25 dpa (thermal neutron spectrum).

For core coolant outlet temperatures around 900°C it is envisioned that existing materials can be used, however, the goal of 1000°C, including safe operations under off-normal conditions, will require the development and qualification of new materials. A large corrosion test programme is necessary to optimise both the materials behaviour and the environmental conditions (chemistry guidelines). Dedicated large-scale helium test loops are under construction, capable of simulating normal and off-normal events.

The focus is also on graphite for the in-core structural materials; on high-temperature metallic materials for internals, piping, valves, high-temperature heat exchangers, gas turbine sub components; and, on ceramics and composites for control rod cladding and other specific reactor internals, as well as for intermediate heat exchangers and gas turbine components for very high-temperature conditions.

Components – In conjunction with the above materials development, design and construction methodologies need to be addressed for key reactor system and energy conversion components (Brayton cycles). These components will require advances in modular manufacturing and site construction techniques, including new welding and post-weld heat treatment techniques.

5. Hydrogen production technologies and nuclear process heat applications (HP)

Co-generation of heat and power makes the VHTR an attractive heat source for large industrial complexes such as refineries and petrochemical industries to substitute large amounts of process heat at different temperatures. This includes hydrogen generation for upgrading heavy and sour crude oil, and process heat applications for metallurgical processes and the production of synthetic hydrocarbon fuels.

The two main hydrogen production processes without GHG emissions are

- thermo-chemistry or "*thermo-chemical water splitting cycles* (TCWSC)" out of which the sulfur/iodine (S/I) thermo-chemical cycle seems to be the most promising
- electro-chemistry or "*high-temperature electrolysis of steam* (HTES)".

Performance and optimization of both processes will be assessed through integrated test loops, from laboratory scale to pilot scale before constructing a demonstration scale prototype and include component development such as advanced process heat exchangers. The NGNP is the US demo plant of VHTR, designed to produce hydrogen and electricity: outlet T of 1000 °C; modules of 600 MWth; 50 % efficiency; capability to produce up to 200 MT of hydrogen per day, the equivalent of 750 000 litres of gasoline; expected to be built by 2017.

A risk analysis will be conducted to limit, as far as possible, the interfacing events between nuclear and non-nuclear plants, in particular, hydrogen explosion, tritium permeation and thermal disturbance caused by the hydrogen production system.

6. Helium Turbine (turbo-machinery) and BOP (project shared with GFR)

The challenges in the power conversion system are related, in particular, to the long-term option of temperatures beyond 1000 °C (e.g. maintenance interval of 60 000 h). This approach will need either cooling of the high-temperature part of the turbine or new (ceramic) materials allowing for higher operational temperatures over long lifetimes. A large-scale test loop will be needed for the entire power conversion system (turbo machinery; compressor; magnetic bearings).

As of the end of 2007, the main contribution of Euratom to the GIF collaborative scheme was the FP-6 Project "RAPHAEL" (*ReActor for Process heat, Hydrogen And ELectricity generation*), an Integrated Project, funded for 4 years with a total budget of 20 M EUR including 9 M from the EC, initiated in April 2005 and co-ordinated by AREVA NP GmbH Erlangen [31] and by the HTR technology network [32] (secretariat at JRC/IE Petten).

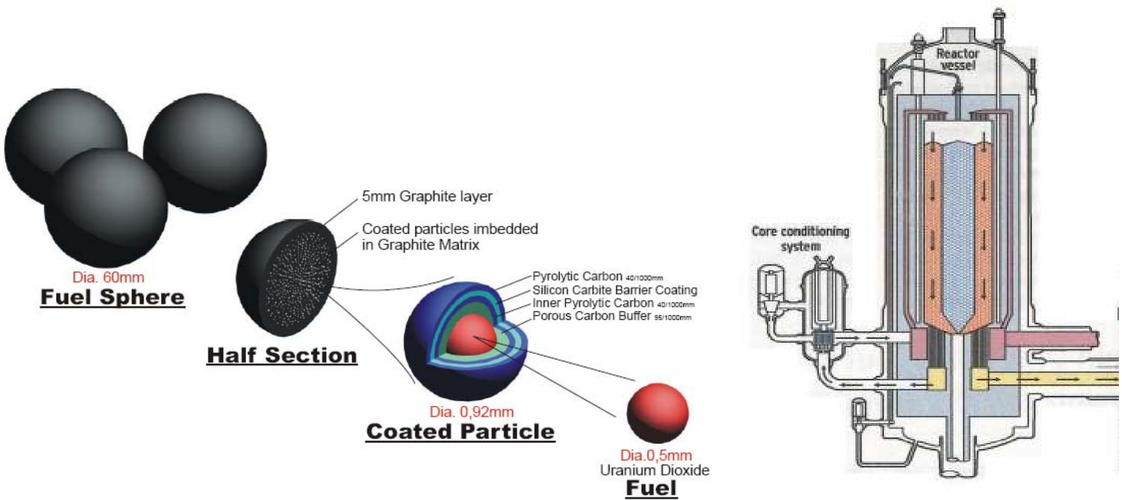

Figure 7 - *VHTR: "how to make Hydrogen out of Helium"* (graphite spheres containing coated fuel particles *left* / PBMR with fuel supply and extraction process *right*)

[31] http://www.raphael-project.org/index.html

[32] https://odin.jrc.nl/htr-tn/

8 SODIUM-COOLED FAST REACTORS (SFR) / ELECTRICITY PRODUCTION AND FULL ACTINIDE MANAGEMENT (ENHANCED FUEL UTILISATION)

The sodium-cooled fast reactor (SFR) system uses liquid sodium as the reactor coolant, allowing for high power density with low coolant volume fraction – Figure 8. The SFR system features a closed fuel recycling system, thereby complying with the GIF technology goal of sustainability. The Generation IV Technology Roadmap concluded that the primary mission for the SFR should be management of high-level wastes and, in particular, management of plutonium and other actinides (burning or breeding). With innovations to reduce capital cost, the mission can extend to electricity production with high efficiency, given the proven capability of sodium reactors to utilize almost all of the energy in the natural (and even depleted) uranium, versus the less than 1% of the potential energy utilized in the natural uranium in thermal spectrum systems. Process heat applications can also be addressed. Owing to the significant past experience accumulated with SFRs in several countries, the deployment of SFR systems is targeted for 2020, the date for a prototype of 250 – 500 MWe.

SFR specific research plans have been arranged into 5 Project Arrangements:

1. System Integration and Assessment (SI&A)
2. Safety Review of Design Options (SO)
3. Advanced Fuel (AF)
4. Component Design and BOP (CDBOP)
5. Global Actinide Cycle International Demonstration (GACID) (Monju reactor)

The basic technology for the SFR has been established in former sodium fast reactors over five decades and in eight countries worldwide. The SFR system builds on more than 300 reactor-years experience (Section 3). The Generation IV version of the SFR, however, should demonstrate that it can effectively manage all actinide elements in the fuel cycle, including uranium, plutonium, and minor actinides (neptunium, americium and curium).

MAJOR CHALLENGES OF THE 5 R&D PROJECTS RELATED TO SFR

1. System Integration and Assessment (SI&A)

Three reference options are available (loop-type, pool-type and modular-type systems, all with a power density of circa 200 $MWth/m^3$):

- A large size (600 to 1500 MWe) <u>loop-type</u> sodium-cooled reactor with mixed U-Pu oxide fuel, supported by a fuel cycle based upon advanced aqueous processing at a central location serving a number of reactors – in the “loop” design, the primary vessel contains the core and feeds, whereas the intermediate heat exchangers and the primary sodium pumps are external (as in Rapsodie, SNR-300, Joyo and Monju)
- A medium or large size (600 to 1500 MWe) <u>pool-type</u> system also supported by a fuel cycle based upon advanced aqueous processing – in the “pool” design, all of the primary sodium stays in the primary vessel which contains the core as well as the intermediate heat exchangers and the primary sodium pumps (as in Phenix, Superphenix, PFR, BN-600 and EFR).
- A small size (50 to 150 MWe) <u>modular-type</u> sodium-cooled reactor with uranium-plutonium-minor-actinide-zirconium metal alloy fuel, supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.

The Generation IV SFR is thus an evolutionary development of the first generation of SFRs, with the following characteristics:

- High potential to operate with a high conversion fast spectrum core (breeding ratio of up to 1.3) with the resulting benefits of increasing the utilization of fuel resources.
- Capability of efficient and nearly complete consumption of trans-uranium (TRU) as fuel, thus reducing the actinide loadings in the high-level waste with benefits in disposal requirements and potentially in non-proliferation.
- High level of safety obtained with the use of inherent and passive means that allow accommodation of transients and bounding events.
- Enhanced economics achieved with the use of high burn-up fuels, fuel cycle benefits (e.g. regarding disposal), reduction in power plant capital costs with the use of advanced materials and innovative design options, and lower operating costs achieved with improved operations and maintenance.

2. Safety Review of Design Options (SO)

In all options, the SFR has a coolant core outlet temperature of 500 - 550°C (core inlet T of circa 370 °C, efficiency of circa 40 %) and a system pressure of 0.1 MPa enabling electricity generation via a secondary sodium circuit. While the oxygen-free environment prevents corrosion, sodium reacts chemically with air, and with water. As a consequence, the design must limit the potential for such reactions and their consequences. Important safety features of the system include a large thermal inertia, a large margin to coolant boiling (boiling point at 0.1 MPa is 882 °C), a primary system that operates near atmospheric pressure and an intermediate sodium system between the radioactive sodium in the primary system and the water and steam in the water plant.

Safety options are investigated through experiments and analytical model developments that cover safety system architectures related to severe accident issues (including passive systems, wherever justified). Of particular interest was the IFR test of inherent passive safety (EBR II, Idaho, 3 April 1986): reactor shut-down without operator or mechanical intervention. Operation options are investigated through testing campaigns, to be conducted in existing reactors (e.g. Monju and Phenix).

3. Advanced Fuel (AF)

Four options are being considered for the SFR fuel: oxide, metal, nitride and carbide. The international research in SFR focuses on the following *fuels* and *claddings*:

- oxide "MOX (TRU bearing)" for the 1500 MWe option (burn up target of 150 GWD/mtHM) and cladding in Oxide Dispersion Strengthened Steel (ODS)
- metal alloy "(U – TRU – 10 % Zr)" for the 50 MWe option (burn up of 80 GWD/t) and cladding in ferritic-martensitic stainless steel (e.g. HT-9 with 12 % Cr, or ODS).

A closed fuel cycle is necessary in order to optimise sustainability (that is: fuel utilization and waste management). Two fuel cycle technology options have been selected:

- an advanced aqueous process (facility located centrally and serving many reactors, appropriate for the large size reactors with MOX fuel)
- the pyrometallurgical process (facilities integrated with the reactors, appropriate for the small size reactors with "U-TRU-Zr" metal alloy fuel).

Both processes have similar objectives：

- very high recovery factor for the management of the actinides (> 99%)
- inherently low decontamination factor, making it highly radioactive
- never separating pure plutonium at any stage.

It is worth recalling that General Electric developed the modular liquid sodium-cooled fast reactor called Super-PRISM ("*Power Reactor Inherently Safe Module*") under the Advanced Liquid Metal Reactor (ALMR) programme in the USA in the late 1990s. This concept uses a dry pyro-processing system that does not separate plutonium from minor actinides, thus enhancing the proliferation resistance of the fuel cycle. This technology gained an increased interest recently in the framework of the *Global Nuclear Energy Partnership* (GNEP initiative, launched in February 2006 by US-DOE – see Section 2).

Various fuel irradiation tests are ongoing with the aim to select advanced fuel options that contain minor actinides and possibly trace fission products. Reactors available for those irradiation tests include BOR-60 in Russia, Phenix in France and Joyo in Japan – Figure 9. Fuel performance codes are being developed, based on available information from previous tests including fuel property measurements and irradiation tests. The fuel evaluation covers both the performance and the fabrication of minor-actinide-bearing fuel (using, in particular, remote techniques) as well as their high burn-up capability.

4. Component Design and BOP (CDBOP)

The main challenges for the structural materials of the two main SFR technologies (pool / loop of 1500 MWe) are related to mechanical integrity:

- first barrier (clad) and sub-assembly (wrapper) under all types of loading (in particular, irradiation induced swelling, creep and embrittlement; thermal creep; fuel – clad interaction) and all types of operating conditions (in particular, normal temperatures between 390 and 700 °C and accidental temperatures up to 850 °C during several hours; "displacement per atom" doses up to 200 dpa)
- second barrier (main vessel and core structures) under various types of thermal and dynamic loadings (e.g. vibrations and accidents) under various conditions (circa 400 °C below and 550 °C above; 1 dpa; helium production, duration of 40 years)
- primary and secondary circuits, and steam generator tubes (primary sodium a 390 – 550 °C and secondary sodium at 330 – 520 °C).

Experimental and analytical evaluation of advanced in-service inspection and repair technologies including leak-before-break assessment are being carried out. For example, a programme of sodium tests with external ultrasonic sensors is being conducted in France. A feasibility study of alternative concepts for energy conversion is conducted in another country. The United States are contributing results of compact heat exchanger tests for super-critical CO_2 Brayton cycle. Japan is providing results, focusing on thermal-hydraulics, liquid sodium/CO_2 reaction, and material corrosion under supercritical CO_2 flow.

5. Global Actinide Cycle International Demonstration (GACID) (Monju reactor)

The joint activities within the GACID project focus on the evaluation of minor-actinide-bearing fuel material property, and analysis and evaluation of irradiated-fuel data, using existing fast reactors. It is a tri-partite irradiation programme (CEA, US/DOE, JAEA), performed in the Japanese reactors Joyo and Monju as a demonstration, on a significant scale, that fast neutron reactors can indeed manage the whole actinide inventory and that the associated technologies comply with the GEN-IV criteria. This project rests on MA bearing fuel material property measurement (high Am content fuel and Cm-bearing fuel). The composition of the GACID fuel is: $(U_{0.674}Pu_{0.25}Np_{0.03}Am_{0.04}Cm_{0.006})O_{2-x}$. Japan has also provided results from previous irradiation tests in Joyo (e.g. Am-1 test).

Here is the experimental test planning:
- Step 1: preparation of the limited minor-actinide bearing fuel preparatory irradiation test (2007-2012) (most likely, only Np-237 and Am-241)
- Step 2: preparation of the licensing of the pin-scale curium bearing fuel irradiation test (2007~2012) (radiation and heat generation issues, most likely Am-243 and Cm-244)
- Step 3: programme planning of the bundle-scale minor-actinide bearing fuel irradiation demonstration (after 2012) (including most likely TRU extraction technology).

As of the end of 2007, the main contribution of Euratom to the GIF collaborative scheme was the FP-6 contract "EISOFAR" [33] (*Roadmap for a European Innovative Sodium Cooled Fast Reactor*), a Specific Support Action, funded for 1 year with a total budget of 0.5 M EUR including 0.25 M from the EC, initiated in January 2007, co-ordinated by CEA Cadarache).

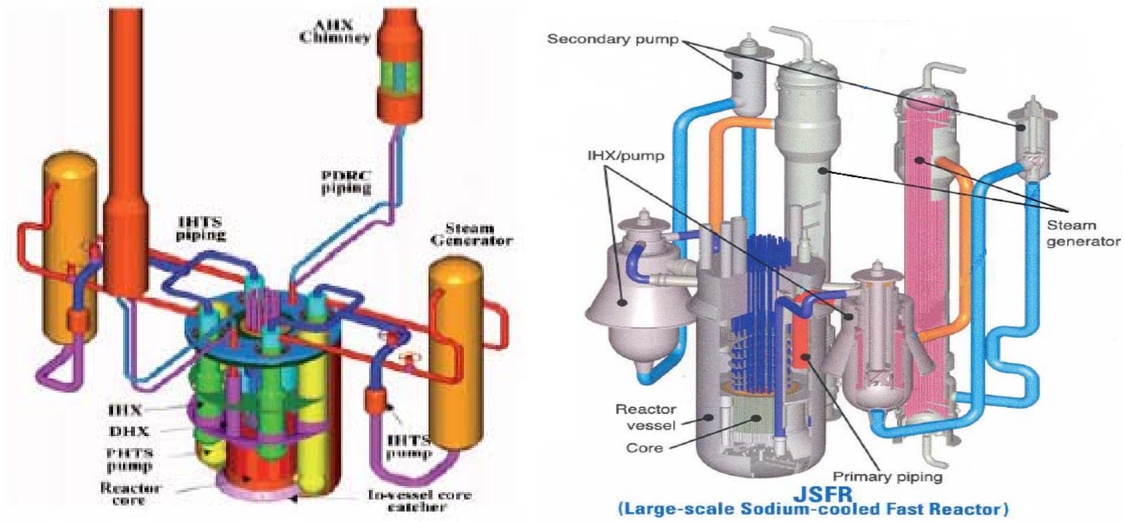

Figure 8 - *SFR: an enhanced sodium cooled fast neutron system* with simplified primary system architecture and full actinide management (pool design *left* / loop design *right*)

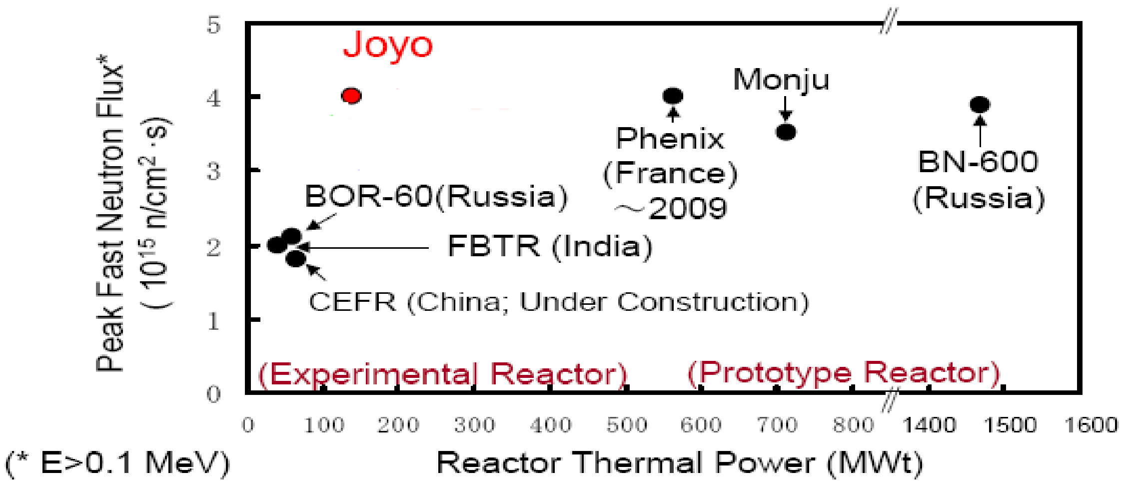

Figure 9 – Worldwide comparison of fast neutron flux facilities (sodium cooled reactors)

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[33] http://cordis.europa.eu/fp6-euratom/lib_projects.htm

9 GAS-COOLED FAST REACTORS (GFR) / COGENERATION OF ELECTRICITY AND HEAT (ENHANCED FUEL UTILISATION)

The Gas-cooled Fast Reactor (GFR) system features a high-temperature helium cooled fast spectrum reactor with a self-sufficient core. Full actinide management is possible, in association with an integrated on-site fuel reprocessing facility. The mission of the GFR system is primarily sustainability in the class of very high-temperature reactors. The high operating temperatures enable high efficiency electricity production (close to 45 %, e.g. through energy conversion with a Brayton gas turbine) but require robust refractory materials. This direct cycle concept is similar to the PBMR and GT-MHR designs (Section 7). The primary difference of these designs is that the GFR would be a breeder. The GFR thus associates the advantages of fast neutron spectrum and high-temperature systems (Figure 10).

A first experimental reactor will be needed in the performance phase to qualify key technologies. This is the Experimental Technology Demonstration Reactor (ETDR). It will be a small power reactor (20 to 50 MWth) with the objective to demonstrate the viability and the performance of specific technologies (e.g. fuel, fuel sub-assemblies, safety systems) and to demonstrate elements of the gas-cooled reactors strategy (decision expected by 2012). It could be put into operation by 2020.

GFR specific research plans have been arranged into 5 Project Arrangements:

1. System Integration and Assessment (SI&A)
2. Fuel, Core Materials, and Fuel Cycle Processes (FCMFC)
3. Other Materials and Generic Technology (shared with VHTR)
4. Components and High-Performance Power Conversion (shared with VHTR)
5. ETDR Integration, Design and Safety (DSM).

Neither experimental reactors nor prototypes of the GFR system have been built so far. Some preliminary GCFR designs, however, were proposed in the 1970s, generally coupled to a Rankine steam cycle. Brayton cycle versions were less well studied. Furthermore, most of those GCFRs employed a *Prestressed Concrete Reactor Vessel* (PCRV), and their helium power conversion units were accommodated inside the PCRV in a direct cycle arrangement. Hence their precedent is of limited value for our present interests. As a consequence, the GFR approach is to rely, as much as possible, on technologies already used (in particular, VHTR).

MAJOR CHALLENGES OF THE 5 R&D PROJECTS RELATED TO GFR

1. System Integration and Assessment (SI&A)
One reference option has been selected. A 2400 MWth unit power is chosen for the pre-conceptual design phase (power density of in the range of 100 MWth/m^3). The choice of a high power for the GFR is dictated by the need for a decreased neutron leakage, which enables the design of self-sustainable cores with not too challenging fuels. The goals are ambitious, including high efficiency, self-sustainable core, and design compatible with long-term decay-heat removal by natural circulation. The reference option is based on ceramic plate type fuel and direct cycle whereas the back-up option is based on ceramic pin type fuel, indirect super-critical CO_2 and He/N_2 combined cycle.

For reasons connected to the construction and operation of the above ETDR, a low power core, however, is also investigated. The 600 MWth GFR concept option (with a direct /

indirect cycle comparison) has been selected for the Euratom project GCFR (see below) with the other Generation IV GFR members taking complementary options.

The main technical challenges that must be addressed to demonstrate the viability of the GFR are as follows:

- core design, achieving a fast-neutron spectrum for effective conversion with no fertile blankets (blanket free of fertile uranium to reduce the proliferation risk) and a gas voiding reactivity effect naturally not significant in the core
- development of core structural materials with superior resistance to fast-neutron fluence (> 100 dpa) under very high-temperature conditions (operational temperatures of helium in the GFR core: 450 °C at inlet and 850 at outlet under 7 MPa
- safety, including decay heat removal systems (in particular, after depressurisation faults), that address the particularly demanding challenges of high power density and reduced thermal inertia (elimination of graphite moderator typical of thermal reactors)
- favorable economics owing to a high power conversion ratio and to the possible coupling with process heat applications
- effectiveness of the whole process regarding generation of solid, liquid and gaseous effluents and the potential for integration with fuel manufacture and on-site treatment.

As far as decay heat removal is concerned (hypothetical severe accident conditions), cores should be designed to achieve a low pressure drop value in order to facilitate coolant circulation in any situation. A medium containment pressure safety strategy is chosen for the decay-heat removal: this means that the design has to include a guard vessel capable of maintaining a pressure of 0.6 to 1.0 MPa in case of primary circuit failure. The rationale for this is to establish compatibility with low pumping power for the emergency gas circulation and to offer the possibility to rely on natural circulation only after several hours after shut-down. Safety research also focuses on the management of anticipated severe accidents and other extreme situations that affect the core, instrumentation and control.

2. Fuel, Core Materials, and Fuel Cycle Processes (FCMFC)

The main technical challenges for the fuel and fuel cycle processes are:

- fuel compositions and forms for the fast-neutron spectrum and high-temperature (the target temperature for GCFR is 850 °C, to be compared with the 550 °C for liquid metal fast reactor cores) - need for refractory materials in the fuel
- self-generation of plutonium in the core to ensure uranium resource saving (converting the natural uranium fuel in the core to fissile material / self-breeding cores)
- ability to transmute long-lived nuclear waste resulting from the recycling of spent fuel, without lowering the overall performance of the system, that is: minor actinides (in particular, Am-241) and long-lived fission products are present in the fuel element
- fuel cycle technology, including simple and compact spent-fuel treatment (e.g. advanced aqueous, pyrochemical methods or combinations of the two) and refabrication for recycling (in particular, global recycling of Pu and all MAs).

One of the main problems of Generation IV fuels is the inclusion of minor actinides (a Generation IV prerequisite) in the fuel fabrication process of the fast neutron spectrum systems (SFR, GFR and LFR). The minor actinides are strong radioactive emitters (in particular, the strong alpha and gamma emitter Am-241). Research focuses on innovative fuel compositions that retain the fission products and allow for large amounts of gaseous fission products and helium gas to be accommodated without high internal stress, swelling and pore formation, thereby accommodating the major irradiation damage effects, while being relatively easy to fabricate. Helium is generated by alpha reactions (e.g. Pu-238, Am-241 and Cm-244). The reduced swelling leads to reduced fuel cladding mechanical interaction, which is especially advantageous for enhanced strategies in actinide management and fuel burn-up. Candidate compositions for the MA bearing fuels in the fast spectrum systems include carbides, nitrides, and eventually oxides, as well as composite ceramic fuels (e.g. cermet or cercer) and new fuel configurations (e.g. coated particle compacts or coated plate).

In general, candidate forms for the ***fuels*** follow two logics (also depending on the time horizon), both aimed at minimizing irradiation-induced property changes:

- solid solution fuel concept (pin) in the short term: incremental innovation of the traditional concept of pellets in a pin (e.g. classical strategy of 271 pins in a hexagonal subassembly). The fabrication of nitride and carbide (U,Pu) pellet type fuels (without MAs) was mastered in the past. As the Generation IV criteria require the management of the MAs in the fuel cycle, pellets will be made with a solid solution of (U, Pu, MA)C for GFR (and (U, Pu, MA)N for LFR). Although some experience exists for oxide fuels, little experimental experience exists for Am containing nitride or carbide fuels. For transmutation purposes in ADS, fertile-free (that is: without uranium) fuels are developed, such as (Pu, Am, Cm, Zr)O2-x or (Pu, Am, Cm, Th)O2-x).

- dispersion solution fuel concept (plate) in the longer term: deployment of more radically innovative concepts (confining the fission damage close to the place of the fissile inclusions). In this concept, the fuel is a so-called *Inert Matrix Fuel* (IMF) composed of two phases: fissile material embedded in an inert, i.e. transparent to neutrons, matrix phase. The fuel is made of particles (for GFR: e.g. (U, Pu)C / O_2) or of sticks embedded in an inert matrix, usually a SiC matrix able to resist operating temperatures of 1200 °C and accidental temperatures up to 1600 °C. The fuel element is a coated compact or plate, which in turn is assembled in a block type structure or a plate sub-assembly, e.g. an honeycomb structure hermetically closed, made of improved composites (e.g. fibre reinforced SiCf/ SiC) or refractory metallic alloys.

Innovative ***cladding*** materials will also be developed to resist fast neutron fluence and high-temperatures (500 – 1000 °C), such as: ceramic materials of the Silicon Carbide (SiC) type, refractory alloys or both (cermet fuel - no ZrC coating).

3. Other Materials and Generic Technology (shared with VHTR)

The VHTR and the GFR systems share a common base of technology for the out-of-core structural materials (e.g. Ni based superalloys), vessel (ferritic-martensitic or F-M steel) as well as for the out-of-reactor block components for energy production or conversion. The in-core components of the GFR, however, are made of refractory metals and alloys, ceramics and ODS (no graphite). Besides the above-mentioned refractory fuel element, additional research is needed on design, fabrication (realistic technology) and behavior of many components under extreme conditions of temperature and irradiation. Moreover, generic research and development is also necessary for helium purification systems, tribology and corrosion, tightness, helium rotating seals, thermal insulation, instrumentation and repair.

4. Components and High-Performance Power Conversion (shared with VHTR)

Very advanced options for energy conversion are envisaged, such as the direct He-Brayton cycle and the super critical CO_2 cycle, both associated with capabilities of high efficiency and compactness. More proven technologies are also being looked at, such as the indirect combined cycle using a He/N_2 mixture for the secondary circuit and a Rankine cycle for the tertiary. This latter proven technology is compatible with high-temperature energy conversion that is being studied in the framework of the VHTR system. Specific experiments are being run to test systems for the GFR, such as passive safety systems or direct cycle gas turbine, using large Helium test loops. Comparison is made with an indirect cycle (e.g. super-critical CO_2 power conversion systems).

5. ETDR Integration, Design and Safety (DSM)

The ETDR will mark an essential step in the definition of the GFR safety options that will constitute a first reference. Considering a start of operation in 2020, the ETDR will be an essential step in the decisions to be made for launching a prototype GFR system. The D&SM project aims at developing a coherent system – fuel, reactor, cycle options – with a self generating core, a robust safety approach and an attractive level of power density. In this context, the main goals are:

- study and qualify the behavior under irradiation and at relevant temperatures of a first generation of GFR fuel, and of innovative features for core materials for GFRs
- develop and qualify fuel cycle processes, from the fabrication to the reprocessing of irradiated GFR representative fuel S/As
- qualify structural materials in fast spectrum and in representative GFR environmental conditions (temperature, pressure, gas flow)
- qualify, at full scale, major computer codes (physics, operation) needed for the design and the analysis of operating transients (design basis accidents and beyond)
- assess the economic performance.

As of the end of 2007, the main contribution of Euratom to the GIF collaborative scheme was the FP-6 project "GCFR"[34] (*Gas-cooled Fast Reactor*), a Specific Targeted Research Project, funded for 4 years with a total budget of 3.6 M EUR including 2 M EUR from the EC, initiated in March 2005, co-ordinated by AMEC-NNC, Knutsford, UK.

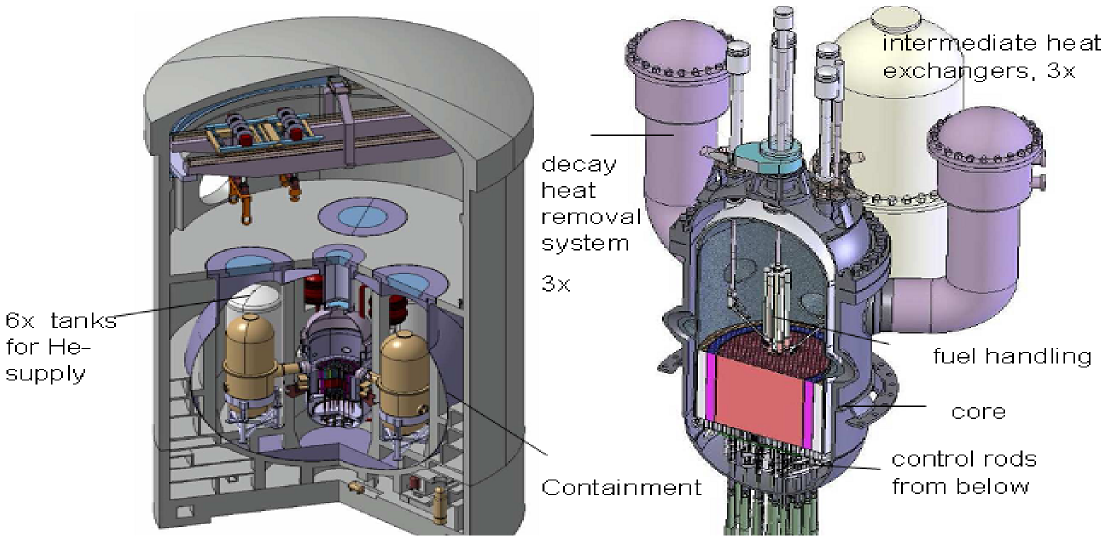

Figure 10 – *GFR: "a VHTR with a fast neutron spectrum"* (that is: MA management)
(no inherent safety features, therefore redundant decay heat removal systems required)

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[34] http://www.gcfr.org/

10 SUPERCRITICAL WATER-COOLED REACTORS (SCWR) / NEXT STEP IN LWR DEVELOPMENT (IMPROVED ECONOMICS FOR ELECTRICITY PRODUCTION)

Supercritical water-cooled reactors (SCWR) are a class of high-temperature, (very) high-pressure water-cooled reactors operating with a direct energy conversion cycle and above the thermodynamic critical point of water (that is: 374°C and 22.1 MPa, single phase coolant for any given heat flux). The supercritical water directly drives the turbine (Rankine steam cycle), without any secondary steam system, much like a BWR. The core inlet and outlet temperatures are 300 and 500°C, resp., under a pressure of 25 MPa. No turbine development is needed as the technology is available in fossil-fired power plants operating at T < 610 °C and P = 25 MPa. Projections of the costs of the SCWR will thus be more accurate than with many developments, since cost models will be based for a large part on proven systems.

The SCWR design is to be the next step in LWR development. It has been proposed to operate at higher temperatures and thermal efficiencies than present LWRs. The reference plant would be 1700 MWe, towards the upper end of present LWR designs. It is built upon two proven technologies:

- LWR and PHWR, which are the most commonly deployed power generating reactors in the world
- commercial supercritical water-cooled (SCW) fossil-fuel-fired (FFP) power plants, which have successfully operated for more than 30 years around the world.

The operation of a 30 to 150 MWe technology demonstration reactor is targeted for 2025.

Compared to conventional water-cooled reactor technologies, the SCWR has key technical advantages that make it attractive as a base load electricity producer in the Generation IV roadmap. The main advantages are improved economics because of

- higher thermodynamic efficiency (efficiency close to 45 %, about one third higher than today's LWRs)
- plant simplification opportunities due to a high-temperature, single-phase coolant, taking also advantage of advanced technologies in conventional power systems.

SCWR specific research plans have been arranged into 3 Project Arrangements:

1. System Integration and Assessment (SI&A)
2. Thermal-Hydraulic Phenomena, safety, stability and methods development (TH)
3. Chemistry and Materials (CM).

Most research on the design has been in Japan and in Germany (e.g. AREVA NP GmbH). No *nuclear* reactor, however, has yet been built using supercritical water as the coolant. This raises a number of challenges for the SCWR, e.g. how cladding materials, will react to the combined radiological and thermo-chemical stresses in the supercritical water environment.

MAJOR CHALLENGES OF THE 3 R&D PROJECTS RELATED TO SCWR

1. System Integration and Assessment (SI&A)
The following reference design (and associated *fuel* – *cladding* system) is assumed:

- the core is based on a thermal spectrum: the fuel is UO_2 and the cladding is F-M (9 to 12 wt% Cr), refractory alloy (Ni based) or ODS (same in thermal and fast version)
- the coolant is light water with a core outlet temperature < 625°C to take full advantage of available technology (materials and chemistry in the SCW-FFP industry)

- a direct cycle will be utilized to eliminate the need for steam generators, steam separators and dryers, pressurizers, and recirculation pumps.
- two different core configurations:
  - pressure vessel like LWR (fuel and other components in one large pressure vessel)
  - pressure tube designs like PHWR / CANDU (fuel and other components contained in a large number of "small pressure vessels" called pressure tubes)
- a once-through open fuel cycle based on traditional ceramic UO_2 or mixed UO_2-PuO_2 fuel will be used for the thermal neutron spectrum option.

The R&D needs to assess technical feasibility (e.g. thermo-hydraulics, materials, chemistry, operating conditions) are common to the two above core configurations. Passive safety features, in particular, are similar to those of simplified boiling water reactors.

Longer-term R&D, which could improve the SCWR concept, consists of the following:
- define and validate hydrogen production techniques that can, technically and cost-effectively, be coupled to an SCWR
- develop more advanced fuel and fuel cycles to improve the performances (construction of an in-reactor fuel test loop to qualify the reference fuel design)
- development of a fast-neutron spectrum version of the SCWR with MOX-fuel:
  - core design with negative void reactivity feedback and safety systems capable of handling much higher core power densities than for the thermal system (development of materials resistant to doses up to 30 dpa in thermal and up to 200 dpa in fast)
  - full actinide management based on advanced aqueous reprocessing, to improve in sustainability by optimum use of uranium as a long-term application.

2. Thermal-Hydraulic Phenomena, safety, stability and methods development (TH)

Much of the interest in the SCWR concept arises from the considerable expected increase in efficiency from the use of supercritical water as the coolant (Figure 11). This results from:
- the higher range of operating temperatures that is possible (supercritical water does not change phase in the loop)
- the high specific heat of supercritical water, which enables greater heat transfer per unit volume, thus permitting a lower mass flow rate compared to pressurized water.

From a safety point of view, the main advantage is the use of a single-phase coolant, which prevents any boiling crisis, a serious issue with PWRs, permitting temperatures to be safely raised and avoiding discontinuous heat-transfer regimes within the core. This is also the reason why supercritical water was used as the coolant in SCW-FFPs in the first place.

One of the major challenges, however, is the thermo-physical phenomenology of supercritical water and its impact on materials behaviour, that is:
- the density of supercritical water is a strong function of temperature and pressure (it can vary by a factor of five or more in the operating temperature range of the SCWR)
- the specific heat features a dramatic peak at the critical point itself (factor 10)
- when water becomes supercritical within the loop, any inorganic impurities dissolved in it would be deposited on reactor materials (insolubility of inorganics).

3. Chemistry and Materials (CM)

The feasibility of the SCWR concept will be decided based on whether materials can be found that can withstand the combined thermal, hydrostatic, thermo-chemical and radiative stresses arising from the operating conditions over the lifetime of the reactor. Steam / water above the critical point is highly corrosive and requires special materials design. This is a particular issue for in-core reactor materials, and less so for balance-of-plant materials, which can use materials previously tested in fossil-fired SCW plants.

In general, reactor materials are stressed by four main factors (and their collective impact): (i) High-temperatures; (ii) High pressures; (iii) Thermo-chemical environment; (iv) Irradiation flux. Some subsequent phenomena may limit the operating lifetime of the reactor.

When structures are submitted to high temperatures and pressures, heavy stresses can be induced, such as *creep (*slow plastic deformation of materials under constant stress) which must be properly considered during the plant operation. Also, the combined effects of chemical corrosion and stress on the materials may cause *stress corrosion cracking* (SCC).

When a material is irradiated, the following changes are induced:

- *embrittlement* (resulting in a loss of material elasticity) - two processes are possible:
  - hardening, resulting from microstructural movement of the dislocations, which reduces ductility
  - grain boundary weakening, caused by preferential diffusion of transmutation products, such as helium, or tramp elements, such as phosphorus, to the grain boundary
- *swelling* (isotropic volume expansion of an irradiated material) - at high doses (10 - 100 dpa), swelling may reach several tens of percent of original volume
- *irradiation creep* (slow change in the shape of a material) in response to an applied radiative stress, over and above the ordinary thermal creep.

Water Chemistry

The single most important parameter in the practical operation of the SCWR (with effects on both the reactor core and the BOP) is the chemistry of supercritical water in the presence of radiation. Control of the chemical composition of the coolant water is very important and has in fact been critical to the continued operations of LWRs, just as it will be for SCWRs.

The observed mechanisms for chemical environment-sensitive cracking in LWRs are

- intergranular stress corrosion cracking (IGSCC)
- irradiation assisted stress corrosion cracking (IASCC)
- corrosion fatigue.

IASCC in austenitic stainless steels is more significant above a *radiative fluence threshold* of about 1 dpa. Further, in nickel-based super alloys, IASCC is sensitive to the presence of impurities such as phosphorous, silicon, boron, or sulphur. The concentrations of the transient and stable species depend upon the following factors:

- radiolysis of the water in the presence of radiative flux
- thermal decomposition of the water due to the higher operating temperature.

The pH of the water naturally has a direct impact on the corrosion potential and rate, and to some extent, also on the mode of corrosion. The control of pH, while theoretically possible, may be difficult in practice, especially in the 300 to 650 °C temperature range.

Materials behaviour under high-temperatures and pressures, and irradiation

The main objective of this project is to select key materials for use both in-core and out-core (corrosion and SCC tests), for both the pressure tube and pressure vessel designs, for the service conditions required. Part of the work will require the definition of a reference water chemistry under irradiation, based on materials compatibility and the radiolysis behavior at supercritical conditions. Special efforts will be devoted to the characterisation (e.g. mechanical properties, dimensional and radiation stability) and qualification of materials tests and to the optimisation of the water quality control strategy.

In SCWR structures (that is: cladding; in-core and out-of-core materials) where the temperatures are > 300 °C or irradiation doses are several dpa, the primary candidate structural materials are (see Generation IV Roadmap [28]):

- ferritic-martensitic (F-M) stainless steel alloys (9 – 12 wt % Cr range: intrinsically more swelling resistant than austenitic steels (demonstrated at doses up to 100 dpa)
- austenitic stainless steel alloys with low swelling properties (Fe Cr Ni alloys)
- Oxide Dispersion Strengthened ODS (advanced F-M steels): the cubic-centered structure provides the irradiation swelling resistance while the dispersed oxides (e.g. yttrium oxides) provide enhanced high-temperature strength (up to 650 °C)
- nickel based alloys (e.g. Fe-35 Ni-25 Cr-03 Ti or high-Mo Ni based like INOR-8).

In general, the above materials are good candidates for all structures of all GIF systems. For GIF systems other than SCWR, the following materials are also good candidates:

- refractory metals and alloys (Fe Cr Ni alloys): in GFR, LFR, VHTR and MSR
- ceramics
- graphite: in VHTR and MSR.

As of the end of 2007, the main contribution of Euratom to the GIF collaborative scheme was the FP-6 project: “HPLWR” [35] (*High Performance Light Water Reactor*), a Specific Targeted Research and Training Project, funded for 3.5 years with a total budget of 4.65 M EUR including 2.5 M from the EC, initiated in September 2006, co-ordinated by FZK.

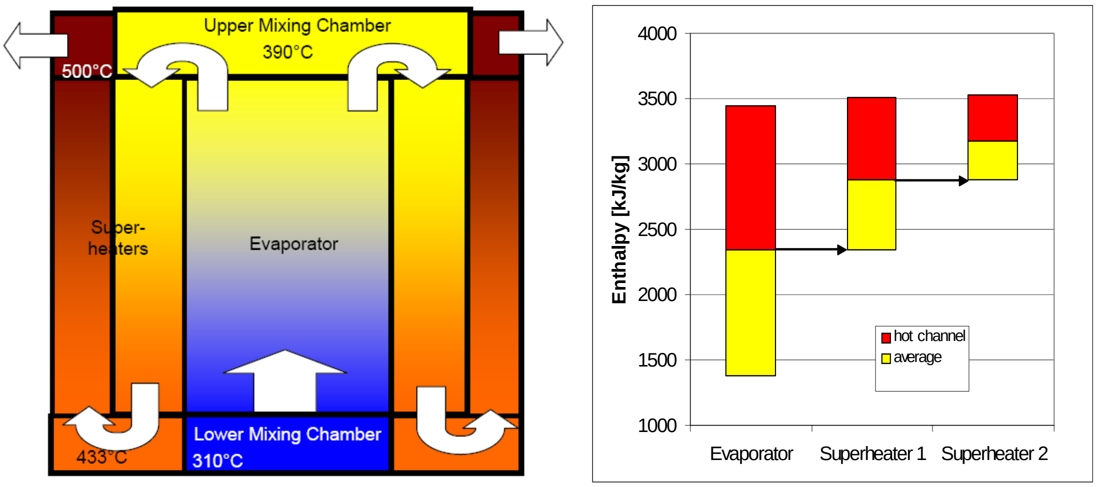

Figure 11 – *SCWR: "a reactor with light water, but supercritical (500°C, 25 MPa)"* (three pass core concept *left* / flow scheme and coolant enthalpy rise *right*)

[35] http://www.hplwr.eu/

11 LEAD-COOLED FAST REACTORS (LFR) / COGENERATION OF HEAT AND ELECTRICITY (FULL ACTINIDE MANAGEMENT, A POSSIBLE SMALL TURNKEY PLANT)

The lead-cooled fast reactor (LFR) system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides (Figure 12). The coolant could be either lead (preferred option), or lead/bismuth eutectic (LBE, back-up solution. Properties of molten lead or lead-bismuth alloy coolants in LFR systems offer potential advantages for reactors with passive safety characteristics (e.g. natural convection), modular deployment, and fuel cycle flexibility. Power target ranges for this reactor would be 10 -100 MWe for batteries or 300 – 600 MWe for central power plants. That would be rather small by historic nuclear standards, but might meet localized market needs. One design favoured under the Generation IV would result in long periods between refuelings, 10 - 15 years.

In addition to realizing those engineering objectives, the feasibility of such systems will rest on development or selection of fuels and materials suitable for use with corrosive lead or lead-bismuth. A full actinide management fuel cycle with central or regional fuel cycle facilities is envisioned.

The *GIF Technology Roadmap* identified the Lead-cooled Fast Reactor (LFR) as a technology with great potential to meet the electricity needs of remote sites as well as for large grid-connected power stations. The LFR system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. The LFR can also, like other fast spectrum reactors be used as a burner of all actinides from spent fuel by using inert matrix fuel. A burner/breeder version could use thorium matrices.

The LFR was primarily envisioned for missions in electricity and hydrogen production, and actinide management. Given its R&D needs for fuel, materials, and corrosion control, the LFR system was estimated likely to afford a technology demonstration by 2025.

LFR specific research plans have been arranged into 3 Project Arrangements:

1. *System Integration and Assessment (SI&A)*
2. *Fuel development*
3. *Lead technology and materials (e.g. primary pumps and steam generator).*

Neither experimental reactors nor prototypes of the LFR system have been built so far. Only Russia has some experience with molten lead or lead-bismuth eutectic (LBE) coolants in nuclear systems. The former Soviet Union used LBE coolant in their “Alfa class” LFR submarines (155 MWth, now decommissioned). The LBE alloy (44.5 % Pb and 55.5 % Bi) has a melting point of 125 °C and a boiling point of 1670 °C at 0.1 Mpa. Difficult steel corrosion problems were solved by careful control of the oxygen content of the circulating molten metal. Potential disadvantages of LBE include the formation of long-lived radioactive lead and bismuth isotopes (lead-205 and bismuth-208 and 210) as well as polonium-210, a strong alpha emitter with a 138 day half-life and a blood cell poison if ingested (remember the death of the Russian ex-spy Alexander Litvinenko in London, November 2006). The Soviet submarine reactors were significantly lighter and more flexible than water-cooled reactors, but lacked reliability, as solidifying of the lead-bismuth solution turned the reactor inoperable.

The above experience nevertheless provides a good basis for the development of new LFR systems, capable of generating electric energy at competitive costs and using plutonium in a closed fuel cycle. The LFR development under the GIF approach is thus to rely, as much as possible, on technologies already used and on the experience acquired, in particular, in

- *Liquid Metal Reactors* (LMR): Superphénix (SPX1) and European Fast Reactor (EFR) programme (sodium coolant) as well as Russian submarine reactors (LBE coolant)
- *Partitioning and Transmutation* (P&T) and *Heavy Liquid Metal* (HLM) technology as part of the research conducted in *Accelerator Driven Systems* (ADS).

One of the aims is to develop a feasible design for a flexible conversion reactor system for time dependent management of actinides. The future plant should have a variety of conversion ratios (CR):

- CR near 0 to transmute legacy waste (no production of fissile nuclei)
- CR near 1 to operate in closed cycle (fissile mass produced = fissile mass destroyed).

As a result of the discussions on the *Energy Amplifier* of Carlo RUBBIA in the late 1990s, the 5th Euratom Framework Programme (1998 – 2002) launched a large ADS research programme, covering partitioning and reprocessing techniques, basic nuclear data, materials behavior, and innovative fuel development for transmutation purposes. The aim was to develop an experimental ADS (XADS). As a fast neutron spectrum is the best choice for transmutation, the R&D efforts focused on both liquid metal-cooled and gas-cooled XADSs. Three sub-critical XADS concepts were investigated (two of them being cooled by LBE):

- an 80 MWth LBE-cooled concept, 110 W/cm, single batch loading
- an 80 MWth Gas-cooled concept, 250 W/cm, single batch loading
- a 50 MWth LBE-cooled concept (MYRRHA, "*Multi-purpose hYbrid Research Reactor for High-tech Applications*"), 500 W/cm, multi- batch loading – Appendix 3.

As a follow-up, under the Euratom FP-6 (2003 - 2006), two large integrated projects were launched: EUROPART (partitioning) and EUROTRANS (transmutation).

The activities of EUROTRANS, which are relevant for the LFR work, are:

- the design of the three previously-described low-power sub-critical XTADS cooled by the Pb-Bi eutectic with a standard MOX driver fuel and of the conceptual design of the higher-power *European Facility for Industrial Transmutation* (EFIT), cooled by pure lead and loaded with U-free transmutation dedicated oxide fuel
- development and qualification of structural materials and HLM technologies (lead and lead-bismuth) for transmutation systems, where the HLM is both the spallation material and the core coolant. This involves specification and fabrication of reference materials, their characterization in the HLM, irradiation studies, thermal-hydraulics, measurement techniques, large-scale integral tests
- improvement of nuclear data bases and models which involves sensitivity analysis and validation of simulation tools, low and intermediate energy nuclear data measurements, nuclear data libraries evaluation at low and medium energies, and high-energy experiments and modelling.

MAJOR CHALLENGES OF THE 3 R&D PROJECTS RELATED TO LFR

1. System Integration and Assessment (SI&A)

Two reactor size options are being considered:

- a small transportable system of 10 to 100 MWe with a very long core life (10 to 15

years), e.g. for distribution of hydrogen and potable water or for electricity on small grids – *Small Secure (or Sealable) Transportable Autonomous Reactor* (SSTAR)
- a medium system of 300 to 600 MWe, intended for central station power generation and waste transmutation (European ELSY project, the reference design for GIF).

The LFR battery is a small factory-built turnkey plant operating on a closed fuel cycle with cassette cores or replaceable reactor modules. The term battery refers to the set of long-life, factory-built cores, not to electrochemical energy cells. Its characteristics are designed to meet market opportunities for electricity and heat production on small grids, and for emerging countries that may not wish to deploy an indigenous fuel cycle infrastructure to support their nuclear energy systems. This system is particularly proliferation-resistant because of the envisioned long life core.

Core inlet and outlet temperatures will be in the range of 400 and 500 °C, resp., under a system pressure close to 0.1 MPa. In this range of temperatures there are many advantages regarding the reduction of corrosion, creep and thermal shocks in transient conditions (efficiency close to 45 %). Temperatures of 800°C are also envisaged with advanced materials: this would enable thermo-chemical hydrogen production. This corresponds to the Russian BREST-300 fast reactor technology (planned in Beloyarsk) which builds on 40 years experience of submarine reactors. In the long term, a large system of 1200 MWe could be envisaged.

2. Fuel development

The *fuel* is metal (UZr), oxide (MOX) or nitride based, containing fertile uranium and minor actinides, with full actinide management from regional or central reprocessing plants. Nitride fuels are generally favoured for LFR use over metal or oxide fuels due to their compatibility with molten lead and lead-bismuth, in addition to their high atomic density and thermal conductivity. The *cladding* will be made of high-Si F-M (Si is there to increase the heat resistance) or ceramics or refractory alloys. Special research actions are devoted (in synergy, whenever possible, with the SFR system) to advanced fuels aiming at burning MAs and LLFPs, from the points of view of fabrication, operation and reprocessing.

Another promising composite concept is the fuel called sphere-pac, also applicable to SFR and GFR as well as SCWR and VHTR. In this concept, kernels or micro spheres (made of U-Pu-MA or Pu-MA or MA alone with particle size similar to HTR but without coating) are produced by a wet fabrication route which facilitates remote and dust-free operation. Sphere-pac fuel is then manufactured by vibratory compacting of these fuel kernels or micro spheres in a suitable cladding. This technique is relatively simple compared to common fuel pellet procedures required for UO_2 or MOX fuel, which are complex, but can be carried out in standard glove boxes with lead screens (Pu is essentially an alpha emitter). The main advantage of this sphere-pac fuel concept is that the interstitial space in sphere-pac fuel allows for gaseous fission products and helium to be accommodated.

3. Lead technology and materials (e.g. primary pumps and steam generator)

Here are some reasons to go for lead:
- high boiling point (1740 °C at 0.1 MPa) and high heat of vaporization (no phase change and no need to pressurize) – pure lead melts at 327 °C
- no chemical reaction with CO_2, air or water
- fast spectrum
- low neutron absorption

The LFR is cooled by natural convection with a reactor outlet coolant temperature of 550 °C, possibly ranging over 800 °C with advanced materials. Temperatures higher than 830 °C are high enough to support thermo-chemical production of hydrogen.

The neutron exposure level to claddings and structures might be up to 200 dpa. Ferritic/martensitic stainless steels, perhaps with silicon and/or oxide-dispersion additions for enhanced coolant compatibility and improved high-temperature strength, might prove sufficient for low-to-moderate-temperature LFRs, but it appears that ceramics or refractory metal alloys will be necessary for higher-temperature LFR systems intended for production of hydrogen energy carriers.

The materials evolution in Lead-Cooled Reactors can be summarized as follows:

- yesterday's materials: stainless steels (SS316, SS304) are subject to a very rapid corrosion in a lead-coolant environment when the temperatures exceed 550°C
- today's materials: T-91 and other advanced alloys are self-passivating alloys in lead-coolant environment, due to the formation of oxide layers (Max. temperature 625-650 °C). Composite coatings are also under development.
- tomorrow's materials: Oxide Dispersion Strengthened (ODS) steels offer an excellent corrosion resistance for very high operational temperatures (Maximum 700 °C)

As of the end of 2007, the main contribution of Euratom to the GIF collaborative scheme was the FP-6 project: "ELSY" [36] (*European Lead-cooled SYstem*), a Specific Targeted Research and Training Project, funded for 3 years with a total budget of 6.5 M EUR including 2.95 M from the EC, initiated in October 2006, co-ordinated by Ansaldo Nucleare, Italy.

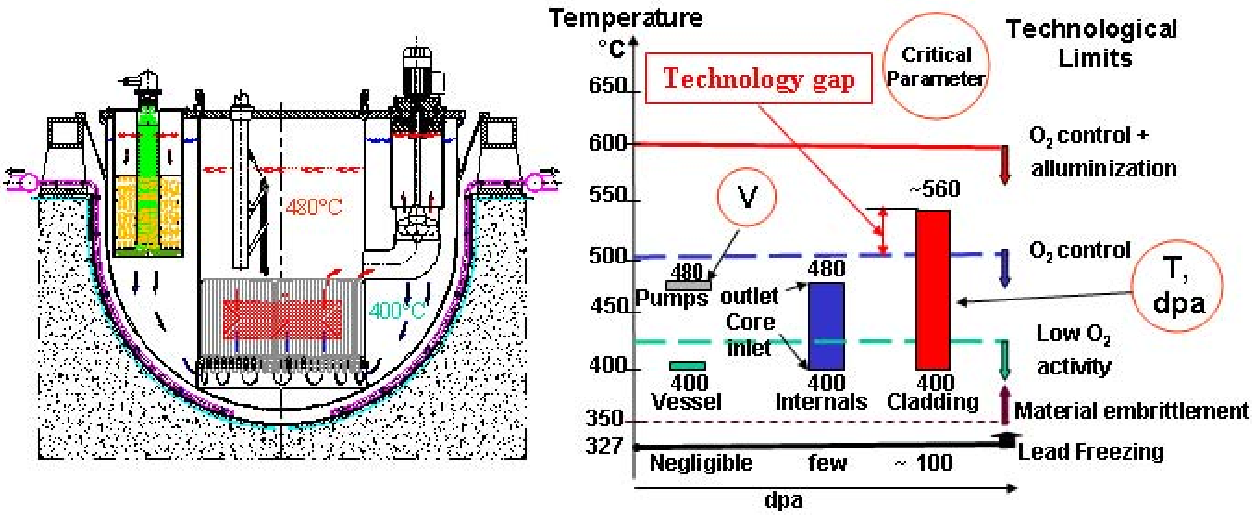

Figure 12 - *LFR: "a turnkey plant with good non-proliferation properties"*/ (ELSY design from ANSALDO *left* / low-temperature thermal cycle to limit corrosion of materials *right*)

[36] http://88.149.184.27/elsywww/

12 MOLTEN SALT REACTORS (MSR) / COGENERATION OF HEAT AND ELECTRICITY (FULL ACTINIDE MANAGEMENT, BREEDING IN THORIUM CYCLE)

The molten-salt reactor system is characterized by the very special feature of a liquid fuel. The fuel is dissolved in a salt coolant, usually fluoride, that is: a circulating liquid of sodium, zirconium, and uranium fluorides as a reactor fuel (Figure 13). All current nuclear power plants use "*solid*" fuel originating sometimes difficulties relating to safety and economy. This problem could be solved by applying the "fluid" fuel concept recommended by Wigner in 1943. MSR systems have a high potential to respond to the criteria assigned by GIF, namely:

- optimum use of fissile materials, with low in-core and fuel cycle inventory along with breeding capabilities (long-term sustainability and improved proliferation resistance)
- minimized production of long-lived wastes, whether configured for burning or breeding (especially in thorium fuel cycle)
- flexible waste management due to high burn-up and on-site reprocessing (no spent nuclear fuel, no requirement to fabricate and handle solid fuel).

The above characteristics may enable MSRs to have potentially unique capabilities and competitive economics for actinide burning and extending fuel resources. The design could use a wide variety of fuel cycles. MSR concepts, which can be used as efficient burners of TRU from spent LWR fuel, have also a breeding capability in any kind of neutron spectrum ranging from thermal (with a thorium based fuel cycle) to fast (with the U-Pu fuel cycle). Temperatures for electricity production would not be as hot as for some other advanced reactors but some process heat potential exists. The initial reference design would be 1000 MWe with a deployment target date of 2025.

MSR specific research plans have been arranged into 5 Project Arrangements:

1. System Integration and Assessment (SI&A)
2. Liquid salt chemistry and properties
3. Fuel and fuel cycle
4. Safety
5. Materials and salt control.

(Epi)thermal neutron spectrum versions of the MSR have been around for some time but were never implemented for commercial uses. The technology was partly developed in the 1950s. As a reminder, the *Aircraft Reactor Experiment*, the first graphite-moderated liquid-fluoride reactor, went critical at Oak Ridge National Laboratory (ORNL, USA) on 3 November 1954. It operated at a maximum temperature of 870 °C and at a maximum power of 2.5 MWth. It also conclusively demonstrated the remarkable chemical and nuclear stability of the liquid-fluoride reactor concept. It was shut down after 100 hours of operation.

The *Molten-Salt Reactor Experiment* (MSRE) programme was then developed as a follow-up by the same ORNL. The MSRE was a 7.4 MWth test reactor simulating the neutronic "kernel" of an inherently safe epithermal thorium breeder reactor. It used three of the reactor fuels: plutonium-239, uranium-235 and uranium-233. Its piping, core vessel and structural components were made from Hastelloy-N and its moderator was a pyrolytic graphite core. It went critical in 1965 and ran for four years. It operated for the equivalent of about 1.5 years of full power operation. The result promised to be a simple, reliable reactor.

The fast neutron spectrum version of MSR was identified as a promising option. The non-moderated thorium-fuelled molten salt breeder reactor (T-MSR-NM) is based on a Th-232 / U-233 closed fuel cycle, has negative feedback coefficients (Dopppler and density) and is predicted to have a doubling time of 30 years. One of the big challenges, however, is the solubility of Pu in the molten fuel salt, fulfilling both chemical and neutronic criteria. Of particular interest is the contribution of Russian research to fast MSRs: the new MOSART concept (*MOlten Salt Actinide Recycler and Transmuter*) is aimed at transmuting Pu and minor actinides without U-Th support. Under the umbrella of the *International Science and Technology Centre* [37] (ISTC), important experimental programmes are being conducted at the Kurchatov Institute on liquid salts (in particular, ISTC Project no 3749, started in 2008).

MAJOR CHALLENGES OF THE 5 R&D PROJECTS RELATED TO MSR

1. System Integration and Assessment (SI&A)

Studies in Europe (national programmes and Euratom FP-5 and FP-6 projects) and worldwide have confirmed the potential of MSR for two major applications:

- breeding in thorium cycle (U-233 / Th, thermal neutron spectrum), like AMSTER
- long life nuclear waste incineration (U – Pu – MA, fast spectrum), like SPHINX.

The AMSTER breeder concept (*Actinide Molten Salt TransmutER,* 2250 MWth, BR > 0.95), proposed by EDF (France), is derived from the MSBR project, but with a drastic release of the constraints on the reprocessing performance. There is no protoactinium (Pa-233) extraction process, which eliminates the major proliferating feature of the concept. A simplified fuel reprocessing scheme is proposed, aimed at minimizing the amount of fuel to be processed, and consequently the amount of actinides sent to the final disposal.

The SPHINX burner concept (*SPent Hot fuel Incineration by Neutron fluX,* 1200 MWth) is proposed by NRI (Rez, Czech Republic): it is a TRU burner in fast spectrum and a radionuclide transmuter in a well-thermalized neutron spectrum.

A rethinking of the safety approach is needed, especially because of the fuel in liquid form and the many chemistry-controlled phenomena. The inlet and outlet temperatures are expected to be in the range of 560 and 700 °C resp. under a system pressure of 0.1 MPa (boiling point at 0.1 MPa is circa 1400 °C). The efficiency will be in the range of 45 %.

Additional advantageous features of the MSR are as follows:

- there is no scenario of the "fuel melt down" type (high level of passive safety inherent to the use of a liquid fuel at atmospheric pressure)
- under accident conditions the fuel is automatically drained into passively cooled critically safe storage tanks
- most gaseous fission products (Krypton, Xenon) are continuously removed so there is no danger of release of these radioactive products, even under accidents conditions
- no fuel fabrication is required, thereby allowing for a wide variety of feeding materials
- high-temperature of the fuel salt is promising for other heat based applications
- several non-proliferation advantages (e.g. simplified fuel logistics)
- small amount of Th resources to generate electricity (with breeding fuel-cycle).

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[37] http://www.istc.ru/

2. Liquid Salt chemistry and properties

Liquid salts offer two potential advantages:

- smaller equipment size because of the higher volumetric heat capacity of the salts
- no chemical exothermal reactions between the reactor, intermediate loop, coolants.

Viability analyses have highlighted the assets of liquid fluoride salts as a coolant for heat transport at high-temperature (700 to 1000 °C) for various applications. As a consequence, new applications for high-temperature heat and new reactor concepts are being developed. The salt coolants have melting points between 350 and 500 ºC and are, therefore, of use only in high-temperature systems. Nitrate salts with a peak operating temperature of around 600 °C are the highest temperature commercial liquid coolant available today. Hence coupling steam cycles to MSRs might be quite complicated because of the need to avoid freezing of the salt. The solution could be the development of closed helium and nitrogen Brayton power cycles.

Research focuses on the following topics, with the aim to select the best salts as fuel and coolant :

- control of the properties of salts (interaction of liquid salts with sodium, water and air)
- clean-up (reprocessing techniques) and flow-sheets (e.g. Th/Ln separation factors)
- development of instrumentation and control; as well as maintenance (e.g. tritium management and control).

The development of higher temperature salts as coolants would thus open new nuclear and non-nuclear applications. These salts are indeed considered not only for intermediate heat transport loops within all types of high-temperature reactor systems (helium- and salt-cooled) but also for hydrogen production concepts, oil refineries, and shale oil processing facilities, amongst other applications. For most of these applications, the heat would have to be moved from hundreds of metres to kilometres.

3. Fuel and fuel cycle

The *fuel* cycle is non conventional: the fuel is dissolved in a fluoride salt coolant, such as alkali fluorides, Th-F_4 based, Zr-F_4 based and fluoroborates. There is thus no such thing like a *cladding*. In the (epi)thermal version, the primary coolant contains also Be-F_2 and the reactor is graphite moderated, as opposed to the fast version. Actually, the Th cycle is possible in two GIF systems: VHTR (Section 7) and MSR. It is usually admitted that better breeding of Th and radiotoxicity are achieved in MSRs. Compared with the standard once-through U fuel cycle in a LWR, a Th fuelled MSR produces 10 – 100 times less MAs (see also Appendix 2).

The MSR technology is actually closely linked to partitioning and transmutation applications. It is a quasi-continuous recycling of fuel in a closed fuel cycle with continuous extraction of fission products, taking advantage of some pyrochemical partitioning techniques. Research focuses on advanced on-line fuel processing, including off gas processing (e.g. bubbling efficiency extraction of gaseous fission products and noble metals). The aim is to simplify fuel reprocessing (preferably on-site: on-line or batch wise), which also means no transportation of spent fuel and no interim storage for spent fuel.

4. Safety

Some of the most recent reactor physics studies focus on conceptual developments for fast neutron spectrum applications, enabling large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors.

Research is also dedicated to other advanced concepts of molten salt systems (some of which were in the original list of the 100 nuclear energy concepts scrutinised by the GIF in 2002 – Section 6). One of the most famous is the *Advanced High-temperature Reactor* (AHTR). It is a reactor concept that combines four existing technologies in a challenging way:

- coated particle graphite-matrix nuclear fuels (used in helium cooled reactors)
- high-temperature Brayton power cycles (used in helium cooled reactors), that eliminate many of the historical challenges in building MSRs
- low pressure liquid (molten) salt coolants with very high boiling points, graphites and specific materials (used in molten salt reactors)
- passive safety systems and plant designs from liquid metal-cooled fast reactors: the better heat transport characteristics of salts compared to helium enable power levels up to 4000 MWth with passive safety systems.

In the USA, the AHTR is viewed as a backup to the helium-cooled VHTR if there are technological or economic factors that limit deployment of the helium-cooled VHTR and as the technological bridge to a MSR. As far as economics are concerned, the AHTRs might also offer the following benefits, compared to helium. Because of the coolant salt properties, it can be built in larger sizes, it operates at lower pressure and the equipment is smaller because of the superior heat transfer capabilities of liquid salt coolants.

5. Materials and salt control

The integrity of components is a big issue, in particular, the mechanical and corrosion behaviour of metallic materials in a high-temperature molten salt environment, the interaction with structural materials and the graphite life durability. Research focuses on the understanding and mastering of fuel and coolant salt technologies, including the compatibility with structural materials, as well as materials development for fuel processing.

As of the end of 2007, the main contribution of Euratom to the GIF collaborative scheme was the Euratom FP-6 project "ALISIA" [33] (*Assessment of liquid salts for innovative applications*), a Specific Support Action, funded for 1 year with a total budget of 0.5 M EUR including 0.25 M from the EC, initiated in February 2007, co-ordinated by CEA Saclay.

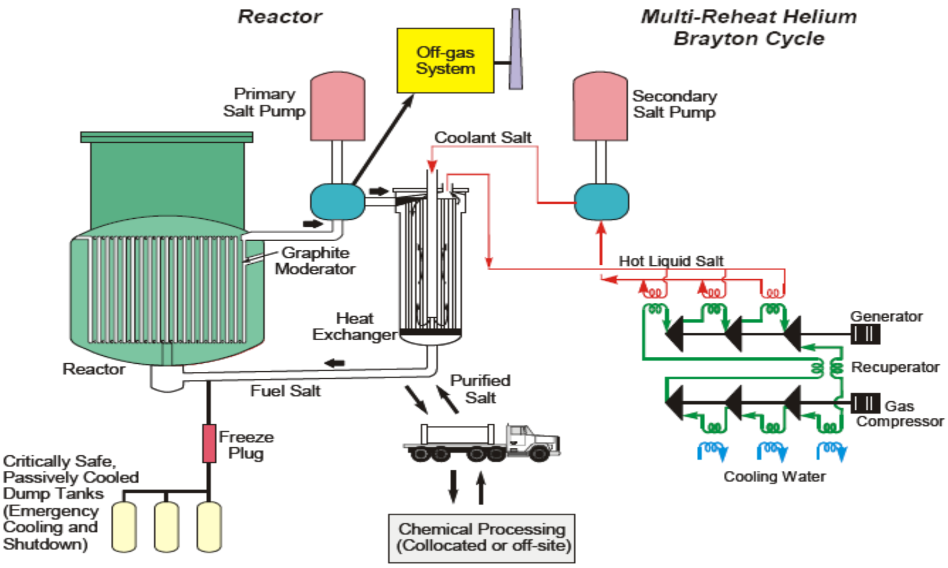

Figure 13 – *MSR: a "2 in 1" system (electricity generating reactor and its reprocessing plant in one and the same site)* (source: C. W. Forsberg, ORNL, Global 2007 Conference)

Gen IV Systems	Electricity Generation	Hydrogen Production & Cogeneration	Actinide Management (AM) Recycle Process	Rating (MWe)	Net efficiency %	Earliest delivery (prototype)
VHTR	▲	▲ ▲ ▲	Once Through	600 MWth (300 MWe)	45 – 50	2020
SFR	▲ ▲ ▲	▲	AM Pyro/Aqueous	50 - 150 600 – 1500 (GNEP)	40	2020
GFR	▲ ▲	▲ ▲	AM Aqueous (on-site)	1000 (ETDR = 50 MWth)	45	2020
SCWR	▲ ▲ ▲	▲	Once Through or AM (Fast) Aqueous	1700	45	2025
LFR	▲ ▲	▲ ▲	AM Pyro (regional)	10 – 100 300 - 600	45	2025
MSR	▲ ▲	▲ ▲	AM Salt Process	1000	45	2025

AM = actinide management

Figure 14 – industrial applications of Generation IV reactor systems (cogeneration of heat & power, plus full actinide management)

Reactor Type	Inlet Temp (°C)	Outlet Temp (°C)	Neutron exposure Max dose (dpa)	Pressure (MPa)	Coolant	Moderator
PWR	290	320	< 100	Relatively high 16	Water	water
VHTR	600	1000	< 25	High 7	Helium	Graphite pebble or Hex block
SFR	370	550	< 200	Low 0.1	Sodium (Teb = 882 °C)	none
GFR	450	850	< 100	High 7	Helium/SC CO_2	none
SCWR	290	510 - 550	< 30 (thermal) < 200 (fast)	Very high 25	Supercritical water	Water rods
LFR	400	550	< 200	Low 0.1	Lead or eutectic (Teb = 1740)	none
MSR	560	700		Low 0.1	Molten salts (fluorides) (Teb = 1400 °C)	Graphite
Fusion (DEMO blanket)	300 (480)	480 (700)	< 100 + helium	8 (0.1)	Helium (83 Pb 17Li)	-

Figure 15 – environmental challenges for Generation IV fuels and structures (in- and outlet core temperatures, neutron exposure, system's pressure)

System	FUEL MATERIALS					STRUCTURAL MATERIALS						
	Oxide	Metal	Nitride	Carbide	Fluoride (liquid)	Ferritic Martensitic Stainless steel alloys	Austenitic Stainless steel alloys	Oxide Dispersion Strengthened steels	Ni-based alloys	Graphite	Refractory alloys	Ceramics
VHTR	P					S	-	-	P	P	S	P
SFR	P	P				P	P	P	-	-	-	-
GFR			S	P		P	P	P	P	-	P	P
SCWR Thermal	P					P	P	S	S	-	-	-
SCWR Fast	P	S				P	P	S	S			
LFR		S	P			P	P	S	-	-	S	S
MSR					P	-	-	-	P	P	S	S

P = Primary option, S = Secondary option
From Generation-IV roadmap

Figure 16 – selection of material classes for fuels and structures in GIF systems

CONCLUSION / IMPACT OF NUCLEAR INNOVATION ON SCIENCE, INDUSTRY AND SOCIETY

In this overview paper, the following question was addressed:

1) What are the main drivers for innovation in reactor systems and fuel cycles at the horizon 2030 ? (Technology Goals of the "Generation IV International Forum" /GIF)

Euratom research in innovative reactor systems and nuclear fuels is presented in the broader context of the key sustainability challenges that drive the EU policy, in particular, the one on "*Climate Change and Clean Energy*". In view of the future "*low-carbon society*", the Energy Policy of the EU relies in part on technological innovation and on the support of appropriate research and development programmes (Section 1). The Presidency Conclusions of the March 2007 European Council reiterated the established position that it is for each EU Member State to decide whether to use nuclear power. Nuclear fission is naturally part of the subsequent *European Strategic Energy Technology Plan* (SET Plan) that covers all primary and secondary energy sources. As a result, a "*Nuclear Fission Initiative*" was proposed, along with five other *Industrial Initiatives*, by the (Energy) European Council of February 2008.

Community research in fission and radiation protection involves a large number of stakeholders (most of them participate in the Euratom programmes – Section 2), such as:

- the nuclear research organisations (public and private, power & medical applications)
- the systems suppliers (e.g. nuclear vendors, engineering companies, etc)
- the energy providers (e.g. electric utilities, heat and/or hydrogen vendors, etc)
- the regulatory bodies and associated technical safety organisations (TSO)
- the education and training (E&T) institutions, and, in particular, the universities
- the civil society and the international institutional framework (IAEA and OECD).

The history of the nuclear fission power technology is recalled in Section 3, namely:

- *Generation I (1950 – 1970): Atoms-for-P*eace era plants (originally from naval designs to prototype commercial power plants, only 4 countries concerned worldwide)
- *Generation II (1970 - 2000):* safety and reliability (expansion to 30 countries worldwide), and pioneering steps in sustainability and efficiency (e.g. HTR and FBR)
- *Generation III (2000 - 2030):* evolutionary steps to further improve safety (severe accident management) and competitiveness (e.g. 60 years lifetime, 90 % capacity).

The answer to the above question lies principally in the four "*Technology Goals for industry and society*" that were proposed by the Generation IV International Forum (GIF, intergovernmental agreement signed by 13 countries worldwide). *Sustainability* goals drive innovation in fuel utilization and waste management (Section 4). *Economics* goals focus on competitive life cycle, energy production costs and financial risk. *Safety and reliability* goals drive innovation in safe and reliable operation, and improved accident management and mitigation of consequences (essentially eliminating the technical need for off-site emergency response). *Proliferation resistance and physical protection* focus on controlling and securing nuclear material and nuclear facilities. The three latter goals are treated in Section 5.

The birth (in 2000) and relatively young history of Generation IV are recalled in Section 6:

- Generation IV (horizon after 2030): visionary innovation regarding sustainability (full actinide management) and competitiveness (e.g. nuclear cogeneration).

The main characteristics of each Generation IV system are discussed in the Sections 7 - 12, following the structure proposed by GIF (that is: several projects for each of the 6 systems):

- *Very high-temperature gas reactors (VHTR)* in Section 7: cogeneration of high-temperature process heat and electricity (efficiency 45 – 50 %) / earliest delivery 2020
- *Sodium cooled fast reactors (SFR)* in Section 8: electricity production and full actinide management (enhanced fuel utilisation/ power 50 - 1500 MWe / earliest delivery 2020
- *Gas-cooled fast reactors (GFR)* in Section 9: cogeneration of electricity and process heat (enhanced fuel utilisation) / full actinide management / earliest delivery 2025
- *Supercritical water cooled reactors (SCWR)* in Section 10: electricity production at high-temperatures (next step in LWR development) / earliest delivery 2025
- *Lead-cooled fast reactors (LFR)* in Section 11: cogeneration of process heat and electricity (full actinide management) / power 10 - 600 MWe / earliest delivery 2025
- *Molten salt reactors (MSR)* in Section 12: cogeneration of process heat and electricity (full actinide management/ breeding in thermal / fast spectrum / earliest delivery 2025.

In this overview paper, another question was addressed:

2) What kind of response is offered by the international RD&DD programmes in nuclear fission (in particular, Euratom) ? (impact on science, industry and society)

The international community is committed to conduct collaborative research programmes under the above-mentioned GIF agreement. The Euratom community, in particular, being one of the GIF signatories, contributes through the indirect (DG RTD) and direct (DG JRC) actions. Not only "applied" research (typical of the Framework Programme, as has been discussed in this paper) but also "fundamental" research is conducted in this context, such as:

- basic science tools to improve experimental research (e.g. laboratory testing)
- basic actinide sciences to optimise the management of long-lived radioactive waste
- mathematical modelling / optimisation of thermo- and electro-chemical processes
- multi-scale numerical simulation of irradiation damage effects on fuels and materials.

The answer to the above question is thus quite obvious. The impact of research of the Generation IV type is quite large not only on science but also on industry and society, namely:

- today, many RD&DD actions (research, development, demonstration and deployment) are conducted, in particular, in the area of structural materials under extreme conditions (good synergy with the Fusion Programme and with non-nuclear research)
- tomorrow, industry and society will benefit from these new power generation technologies, to the extent that the GIF technology goals will be effectively achieved.

In order to structure the Community effort during the current nuclear Renaissance (Generations II and III) and to prepare the way for the future "*low-carbon society*" (Generation IV), a *Sustainable Nuclear Energy Technology Platform* (SNE-TP) was launched in September 2007. The aim is to bring together all stakeholders in the EU-27 and to prepare, in particular, two important guidance documents: the *Strategic Research Agenda* (SRA) and the *Deployment Strategy* (DS). It is clear that the above questions and answers regarding science, industry and society will continue to be at the heart of the SNE-TP debates. It is also clear that concrete actions will be taken (e.g. private – public partnerships) in view of 2012.

APPENDIX 1 – EURATOM RESEARCH AND TRAINING IN THE FRAME OF THE EUROPEAN "HIGHER EDUCATION" (EHEA) AND "RESEARCH" (ERA) AREAS

Definition of Education and Training

For the sake of clarification, Education and Training (E&T) are defined as follows:

- *Education* is a basic or life-long learning process: education is broader than training and encompasses the need to maintain completeness and continuity of competences across generations. It is essentially a knowledge-driven process, involving academic institutions as suppliers, and students as customers.

- *Training* is learning a particular skill required to deliver a particular outcome: training is about schooling activities other than regular academic education schemes. It is essentially an application-driven process, involving "future employers" (or associated training organisations) as suppliers, and professionals as customers.

Towards the "European higher education area": a single umbrella for Education and Training programmes (2007 – 2013) / "ERASMUS" for higher education (DG EAC)

In the EU, education is in principle an exclusive competence of the Member States. Therefore the role of the EU is "limited" to develop the European dimension in education, in particular, by encouraging mobility of students and teachers (e.g. academic recognition of diplomas) and by promoting co-operation between educational establishments.

In this context, the EU launched in 2007 the *Lifelong Learning Programme* (LLP)[38], as a single umbrella to integrate all educational and training initiatives that were originally organised by DG Education and Training (EAC) through the SOCRATES Programme 2000 - 2006. The budget is nearly EUR 7 billion for 2007 to 2013. The LLP enables individuals at all stages of their lives to pursue learning opportunities across Europe. It consists of four "sectoral" sub-programmes:

- **Comenius** addresses the teaching and learning needs of all those in pre-school and school education
- **Erasmus** addresses the teaching and learning needs of staff and students in Higher Education. It also provides support for institutions across Europe to work on shared projects, including curriculum development and other areas
- **Leonardo da Vinci** enables people who are involved in vocational education and training to benefit from work experience placements and career development opportunities in another country
- **Grundtvig** focuses on adult education and funds small-scale, community-based activities.

As regards the four sectoral programmes, quantified targets have been set in order to ensure a significant, identifiable and measurable impact for the programme, namely:

- For **Comenius** (school education): to involve at least three million pupils in joint educational activities, over the period of the programme

[38] http://ec.europa.eu/education/programmes/llp/index_en.html

- For **Erasmus** [39] (higher education): to contribute to the achievement by 2012 of three million individual participants in student mobility under the present programme and its predecessors (reminder: the first ERASMUS programme started in 1987)
- For **Leonardo da Vinci** [40] (vocational training): to increase placements in enterprises to 80 000 per year by the end of the programme
- For **Grundtvig** (adult education): to support the mobility of 7 000 individuals involved in adult education per year, by 2013.

Towards the "European Research Area": a single umbrella for Research and Training programmes (2007 – 2013) / "Marie Curie" actions accompanying research (DG RTD)

In the EU (in particular, in Euratom), training is treated in close connection with research. Therefore the role of the EU is quite important, as has also been underlined in the strategy of the previously mentioned *European Research Area* [2] (ERA, launched in 2000 and revised in 2005). The related EC Communication "*Towards a European Research Area*" proposed ways in which research and training could be more effectively organised and coordinated.

The Seventh Framework Programme (FP-7, EC research over the period 2007 - 2013) has a total worth of more than EUR 50 billion and includes the following Specific Programmes:

- *Cooperation* – fostering collaboration between industry and academia to gain leadership in key technology areas - EUR 32 413 million (64.2 %)
- *Ideas* – supporting basic research at the scientific frontiers (implemented by the *European Research Council* / ERC) - EUR 7 510 million (14.9 %)
- *People* – supporting mobility and career development for researchers both within and outside Europe (follow-up of Marie Curie actions) - EUR 4 750 million (9.1 %)
- *Capacities* – helping develop the capacities that Europe needs to be a thriving knowledge-based economy (support to infrastructures) - EUR 4 097 million (8.3 %)
- *JRC (non-nuclear)* - EUR 1 751 million (3.5 %).

The 7-th Euratom Framework Programme (FP-7, Euratom research over the period 2007 – 2011, five years duration imposed by the Euratom Treaty) consists in:

- *Nuclear research (Euratom programme)* – developing Europe's nuclear fission and fusion capabilities (RTD and JRC) - EUR 2 751 million) – see also FISA conference [41].

The "*Marie Curie Actions*", managed by DG Research (RTD), have long been one of the most popular features of the Community Framework Programmes for Research and Technological Development. They have developed significantly in orientation over time, from a pure mobility fellowships programme to a programme dedicated to stimulating researchers' career development. The "*Marie Curie Actions*" have been particularly successful in responding to the needs of Europe's scientific community in terms of training, mobility and career development.

In the Seventh Framework Programme, the "Marie Curie Actions" have been regrouped and reinforced in the "PEOPLE" Specific Programme [42]. Entirely dedicated to human resources in

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[39] http://ec.europa.eu/education/programmes/llp/erasmus/index_en.html

[40] http://ec.europa.eu/education/programmes/llp/leonardo/index_en.html

[41] http://cordis.europa.eu/FP-6-euratom/ev_fisa2006_proceedings_en.htm

[42] http://cordis.europa.eu/fp7/people/home_en.html

research, this Programme has a significant overall budget of more than EUR 4.7 billion over a seven year period until 2013, which represents a 50% average annual increase over FP-6.

The "PEOPLE" Programme is being implemented through actions under five headings:

- *Initial training of researchers to improve mostly young researchers*
- *Life-long training and career development*
- *Industry-academia partnerships and pathways*
- *International dimension*
- *Specific actions.*

Euratom Education and Training strategy (always connected to Research)
(in close connection with the "future employers", wherever possible)

The goal of the Euratom education strategy is, in collaboration with academia, to develop instruments that help produce top-quality teaching modules that can be assembled into higher level training packages or Masters Programmes that are jointly qualified and mutually recognised across the EU. This is done naturally in line with the above ERASMUS Programme of the LLP. The following four objectives have been agreed upon (ENEN) [43]:

- MODULAR COURSES AND COMMON QUALIFICATION APPROACH
(e.g. ensuring top-quality for each module and developing a coherent framework)
- ONE MUTUAL RECOGNITION SYSTEM ACROSS THE EUROPEAN UNION
(e.g. *European Credit Transfer and accumulation System* of ERASMUS /ECTS/)
- MOBILITY FOR TEACHERS AND STUDENTS ACROSS THE EU
(e.g. broadening the circulation and exchange of ideas and knowledge in nuclear fission)
- FEEDBACK FROM "STAKEHOLDERS" (BOTH PUBLIC AND PRIVATE)
(e.g. involving "future employers" and improving the balance of supply and demand).

The goal of the Euratom training strategy is, in collaboration with "future employers", to identify commonalities amongst CPD actions ("*Continuous Professional Development*"). Areas of common interest are, for example, nuclear safety culture (e.g. ALARA principle), design of Generations III and IV reactors, radiation protection expertise in response to EU Directives (e.g. in hospitals and industry), radiobiology, geological disposal, etc. This is done naturally in line with the above COOPERATION and PEOPLE Programmes of the FP-7.

The following four objectives have been agreed upon with the stakeholders and a new instrument is proposed (EFTS or "*Euratom Fission Training Scheme*") [44]:

- address life-long learning and career development in the areas of Nuclear Fission and Radiation Protection, adopting wherever possible the four above ENEN principles
- maximise transfer of higher level knowledge and technology with emphasis on inter-sectoral mobility across the EU (e.g. internships in the stakeholders' organisations)
- remove all administrative or other barriers to the mutual recognition of professional qualifications in the Internal Market ("*right to provide services anywhere in the EU*")
- ultimate objective = develop "*European Training Passports*" that will enable high-level nuclear experts to exercise in the EU and offer a guarantee of quality to all employers.

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[43] www.enen-assoc.org.

[44] http://cordis.europa.eu/fp7/euratom/home_en.html

APPENDIX 2 - THORIUM BREEDING CYCLE (FERTILE TH-232 => FISSILE U-233)

Thorium is a naturally-occurring, slightly radioactive metal discovered in 1828 by the Swedish chemist Berzelius, who named it after Thor, the Norse god of thunder. Th-232 is actually the most common radioactive element on Earth and has a half life of 14 billion years. It is more common than lead and four times more common than uranium: e.g. monazite sands principally in Australia and India, but also in China and Turkey, and thorite in Norway.

A conversion process is possible with fertile thorium (Th-232 => fissile U-233) in specific thermal neutron reactors. Given a start with some other fissile material (U-235 or Pu-239), a breeding cycle can be set up, without the need of fast spectrum neutrons. As a way of reminder, although not fissile itself, Th-232 may capture slow neutrons to produce protactinium (Pa-233), which decays with a half life of 27 days to U-233, which is fissile (also a potential nuclear weapon material). Compared to other fissile isotopes, U-233 produces best neutron economy (especially in hard thermal spectra).

Th-232 is a better "fertile" material than U-238, because less various non-fissile isotopes are created along the way. Higher fuel conversion ratios and longer fuel burnups are achieved. Thorium dioxide (ThO_2), also called thoria, has the highest melting point of any oxide (3300°C), that is: 500 °C higher than that of UO_2. This property provides an added margin of safety in the event of a power surge or loss of coolant. It also behaves better in-pile because of its higher thermal conductivity and lower coefficient of thermal expansion. It is chemically stable and almost insoluble in groundwater (important for spent fuel management).

In the 1990s, research was conducted on the transmutation of TRU and of fission products within the thorium fuel cycle, following the work initiated in 1993 by Carlo Rubbia. His *"Energy Amplifier"* was aimed at producing energy and burning actinides and fission products, thereby reducing the burden on the back-end of the fuel cycle (secular disposal containing only short-lived wastes).

Today's research focuses more on the breeding than on the burning capacities of the thorium cycle:

- breeding in a fast (non-moderated) neutron flux: *molten salt reactor* with unique possibilities for uranium production, actinide burning and low waste level, avoiding the cumbersome fabrication and multiple recycling of MA in solid fuels
- breeding in a slow (moderated) neutron flux: *light water reactor* using combinations that keep a uranium-rich (typically 20 %) "seed" separate from a thorium-rich "blanket" (e.g. Radkowsky-Kurchatov approach [45] , focusing on non-proliferation of weapons).

The thorium fuel cycle is intrinsically proliferation resistant because it produces much less plutonium and long-lived minor actinides. Another reason is the high radiation fields in the fuel fabrication and thorium recycling processes that are necessary to recover the fissile U-233. Separated U-233 is always contaminated with traces of U-232 (69 years half life, with daughter products such as bismuth-212 and thallium-208 that are strong alpha and gamma-emitters with short half lifes). The spent fuel also contains large amounts of Pu-238 (up to 4

[45] http://www.thoriumpower.com/files/Thorium_Fuel_for_Nuclear_Energy_by_Kazimi.pdf

times more than in the conventional uranium fuel cycle), a highly radioactive isotope, which thus produces a lot of heat.

These properties also have a dramatic impact on the costs of both the fuel fabrication (chemical separation of highly radioactive U-233 from the irradiated thorium fuel) and the recycling process (presence of highly radioactive isotopes). This means more of the process would need to be remotely operated, compared to current plutonium reprocessing which has a lower concentration of highly radioactive materials. As a result, the cost to reprocess is estimated to be 30% greater than for uranium based fuels. During the 1980s, fuel fabrication was on an industrial scale in Germany (in the context of THTR – see below) but nowadays there are no more industrial facilities in Europe.

It is also worth recalling the particular interest of India in the thorium fuel cycle, using another technology than HTR. Their attraction to thorium-based fuels stems, in part, from their large indigenous supply. Since the mid-1990s, they operate two PHWRs of 200 MWe (CANDU type), fuelled with ThO_2-UO_2 containing 500 kg of thorium (Kakrapar-1 and -2).

Pioneering period of the thorium breeding cycle (1960 – 1980) in LWRs and HTRs

The breeding cycle Th-232 => U-233 was first tested in LWRs. During the early 1960s, thorium-based fuels were tested in the USA in LWRs, in particular, at the experimental facility “BORAX” (*BOiling ReActor eXperiment*) in Idaho, for example BORAX IV (a BWR of 20 MWTh / 2.4 MWe, 1963 – 1968). During the next decade, an industrial prototype was constructed, based on combinations of Th-232 and U-235 or U-233, namely: the *Light Water Breeder Reactor* (Shippingport in Pennsylvania, 1977 – 1982). This LWBR of 60 MWe has been the major thorium-based reactor in the world to demonstrate thermal breeding (more fissile material was created than loaded, that is: breeding ratio >1). The reactor operated for nearly 5 years and the fuel achieved a maximum burnup of 60 000 MWd/mtHM without any fuel failure. However, the fuel design produced 30% less power, and was more expensive to manufacture and reprocess than typical uranium fuel.

Later on, during the late 1960s, series of tests for thorium-based fuels were performed successfully in HTRs, which were usually graphite moderated and helium-cooled, and operated with fuel in coated particles / in pebble or compact, such as:

- in Germany (ABB and Siemens Interatom combined to form Hochtemperatur Reaktorbau GmbH): the prototype AVR (“*Arbeitsgemeinschaft Versuchsreaktor*”, 46 MWth, operation 1967 – 1988, overall consumption of more than 1300 kg of thorium, a remarkable burn-up of 150 000 MWd/mtHM) and the power reactor THTR-300 (750 MWth, 1985 – 1989), both operating with pebble beds with ThC_2-UC_2 fuel
- in the USA (General Atomics): prototype of Peach Bottom (115 MWth / 40 MWe, operation 1967 – 1972) and the power reactor of Fort Saint Vrain (330 MWe, operation 1976 – 1989, temperature up to 700 °C, overall consumption of more than 13 000 kg of thorium), both operating with prismatic blocks with ThC_2-UC_2 fuel
- in the UK: experimental reactor DRAGON at Winfrith (20 MWth, OECD-Euratom-UK project, outlet temperature of 850 °C, 1964 – 1973) operating with ThC_2-UC_2 fuel. Dragon was run as an OECD/Euratom cooperation project, involving Austria, Denmark, Sweden, Norway and Switzerland in addition to the UK, from 1964 to 1973. The Th/U fuel was used to 'breed and feed', so that the U-233 formed replaced the U-235 at about the same rate, and fuel could be left in the reactor for about six years.

APPENDIX 3 – TRANSMUTATION OF MINOR ACTINIDES IN ACCELERATOR DRIVEN SYSTEMS (ADS)

Euratom research in sustainable fuel cycles (that is: full management of all actinides resulting from the U/Pu fuel cycle) focuses on the following challenges:

- the substantial reduction of long-term radioactivity, radio-toxicity and fissile materials inventories, thereby improving the public's acceptance of the geological repositories (they are unavoidable: P&T and Generation IV will not eliminate their need)
- the minimization of short- and medium-term heat sources that enable a reduction in the volume required by the high-level wastes in repositories, thereby further increasing their effective capacity and reducing their number.

It is worth recalling that the energy spectrum can be either thermal or fast, depending on the application sought: power generation or destruction of actinides. The fast spectrum (compared to thermal) presents advantages:

- a larger excess of neutrons, created by fission (of Up), which can be used for the generation of Pu-239 from capture on U-238, thereby improving the energetic yield for possible energy production (breeding / enhanced fuel utilisation)
- a higher fission/capture ratio, which results in a smaller production of actinides (Np, Am, Cm) which are also more fissionable, thereby reducing the total amount of minor actinides in the core (transmutation / full actinide management).

As a consequence, transmutation of actinides in fast neutron spectrum facilities can solve the above challenge in a way that is both efficient and sustainable.

Actually, both critical reactors and sub-critical Accelerator Driven Systems (ADS) with a fast neutron spectrum are potential candidates for transmutation. However, in order to maintain a sufficient safety level, critical reactors cannot be loaded with large fractions of MAs due to unfavourable reactivity coefficients and the small delayed neutron fraction. In ADS, high-energy neutrons are produced through the spallation reaction of high-energy protons from an accelerator striking heavy target nuclei that can be directed to a subcritical reactor ($k_{eff} < 1$). The core acts as a neutron amplifier of the primary spallation source. Sub-critical reactors are in fact very efficient and safe to achieve high MA transmutation rates:

- in an ADS, it is possible to sustain a fission reaction which can readily be turned off
- it can be operated in a flexible manner even with a core containing a high amount of MAs.

In 1998, the Research Ministers of France, Italy and Spain decided to establish a Group of Advisors for the definition of a common European R&D platform on ADS. Under the direction of Carlo Rubbia, ex-Director General of CERN (Geneva, Switzerland), an important guidance document was produced in April 2001: *"A European Roadmap for Developing "Accelerator Driven Systems" for Nuclear Waste Incineration"*. One of the recommendations was to design and operate an eXperimental ADS plant (XADS) in a European context.

Among the pioneering projects of international interest in the area of ADS was the *Energy Amplifier* concept, an innovative and, in principle, inherently-safe system, conceived by Carlo Rubbia. Two key experiments were conducted at CERN, partly under the Euratom FP-5 (1998 – 2002), namely: TARC (*Test of Adiabatic Resonance Crossing*, 1993) and FEAT (*First Energy Amplifier Test*, 1994). In this context, a major role was played by SCK•CEN (Mol, Belgium) through their XADS project MYRRHA (started in 1998 – see also Section 11).

MYRRHA / "Multi-purpose hYbrid Research Reactor for High-tech Applications" [46]

The MYRRHA project is a reactor based on the coupling of a proton accelerator with a liquid lead-bismuth-eutectic (LBE) windowless spallation target. The total power of this ADS ranges between 50 to 80 MWth (depending on the core loading and the experimental rigs inserted). The proton accelerator consists of a LINAC of 1.4 MW proton beam heating, with very demanding conditions in terms of beam reliability: less than 5 beam trips per year. The windowless spallation target is surrounded by a Pb-Bi cooled sub-critical neutron multiplying medium in a pool type configuration with a standing vessel. The core maximum sub-criticality level of $k_{eff} \sim 0.950$ assures a comfortable margin for safe operation. The design of the spallation target is the subject of an important international experimental and theoretical programme.

The primary system of MYRRHA is thus made of a pool, containing a fast-spectrum sub-critical core, cooled with LBE for the primary coolant and boiling water as a secondary fluid. The core inlet temperature is 300°C and the outlet temperature is 380°C. The heat exchangers and primary pumps are immerged in the reactor vessel in dedicated casings. The core pool contains several islands housing thermal spectrum regions located in in-pile sections in the fast core. The total neutron flux levels ($1.0\ 10^{15}$ to $5.0\ 10^{15}$ n/cm².s), achieved in large irradiation volumes in the core (about 20 000 cm³ in total), allow very high performance testing conditions. Minor actinide transmutation studies, structural material research and ADS fuel studies can be performed in the fast spectrum zone. Radioisotope production (e.g. for medical applications), long-lived fission product (LLFP) transmutation research as well as LWR fuel safety studies can be conducted in the thermal spectrum island. The MYRRHA fuel design is based, in the first phase, on Fast Reactor MOX fuel technology. MA bearing fuel assemblies will be accepted later on, whenever these will be available. The remote handling for both out- and in-vessel operation and maintenance has been developed on the basis of demonstrated technology in the Joint European Torus (JET) fusion facility.

As a fast spectrum irradiation facility, MYRRHA will thus contribute to the objective of making nuclear fission a sustainable energy source (that is: enhanced fuel utilisation and full actinide management). Besides the reduction of the HLW burden (MAs as well as LLFPs), the MYRRHA project will also contribute to the development of coolants of the lead alloy type, in support to one of the Generation IV reactor systems, namely the Lead Fast Reactor (LFR). Moreover innovative fuels and structural materials of the different Generation IV systems can be tested in safe, well defined and representative irradiation conditions.

The MYRRHA project should confirm that the best transmutation rates for all minor actinides are achieved in ADS using appropriate fuels: it is theoretically in the order of 35 kg / TWhe, i.e. approximately 280 kg of MAs per GWe annual (365 full power days). An ADS of 800 MWth coupled with a 30 MW proton accelerator should be able to transmute 250 kg/year of MAs which corresponds to the amount of MAs produced in 10 LWR units of 1 GWe.

General conclusions about the impact of partitioning, transmutation and waste reduction technologies on the final nuclear waste disposal are discussed in the FP-6 project RED-IMPACT [47], using technical as well as political and societal inputs (public acceptance).

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[46] http://www.sckcen.be/myrrha

[47] http://www.ec.europa.eu/research/energy/fi/fi_cpa/waste/article_2523_en.htm

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