Generation IV Nuclear Reactors

(Updated July 2016)

- **An international task force is developing six nuclear reactor technologies for deployment between 2020 and 2030. Four are fast neutron reactors.**
- **All of these operate at higher temperatures than today's reactors. In particular, four are designated for hydrogen production.**
- **All six systems represent advances in sustainability, economics, safety, reliability and proliferation-resistance.**
- **Europe is pushing ahead with three of the fast reactor designs.**
- **A separate programme set up by regulators aims to develop multinational regulatory standards for Generation IV reactors.**

GIF

The Generation IV International Forum (GIF) was initiated in 2000 and formally chartered in mid 2001. It is an international collective representing governments of 13 countries where nuclear energy is significant now and also seen as vital for the future. Most are committed to joint development of the next generation of nuclear technology. Led by the USA, Argentina, Brazil, Canada, China, France, Japan, Russia, South Korea, South Africa, Switzerland, and the UK are charter members of the GIF, along with the EU (Euratom). Most of these are party to the 2005 Framework Agreement (FA) which formally commits them to participate in the development of one or more Generation IV systems selected by GIF for further R&D. Argentina and Brazil did not sign the FA, and the UK withdrew from it; accordingly, within the GIF, these three are designated as “inactive Members.” Russia formalized its accession to the FA in August 2009 as its tenth member, with Rosatom as implementing agent. In 2011 the 13 members decided to modify and extend the GIF charter indefinitely. In February 2015 the FA was extended for ten years, with Rosatom signing for the extension in June. Australia joined as the 14th member in June 2016.

In 2005 the technical secretariat transferred to the OECD Nuclear Energy Agency (NEA), alongside that of MDEP (see section below), and from mid-2015 also the International Framework for Nuclear Energy Cooperation (IFNEC). The NEA in Paris provides the technical secretariat to support the system steering committees

(SSCs), project management boards (PMBs), modelling working groups (MWGs) and task forces (TFs) for GIF.

GIF focus

After some two years' deliberation and review of about one hundred concepts, GIF (then representing ten countries) late in 2002 announced the selection of six reactor technologies which they believe represent the future shape of nuclear energy. These were selected on the basis of being clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while being resistant to diversion of materials for weapons proliferation and secure from terrorist attacks. They are the subject of further development internationally, with expenditure of about $6 billion over 15 years. About 80% of the cost is being met by the USA, Japan and France.

In addition to selecting these six concepts for deployment between 2010 and 2030, the GIF recognised a number of International Near-Term Deployment advanced reactors available before 2015. (see Advanced Reactors paper)

Most of the six systems employ a closed fuel cycle to maximise the resource base and minimise high-level wastes to be sent to a repository. Three of the six are fast neutron reactors (FNR) and one can be built as a fast reactor, one is described as epithermal, and only two operate with slow neutrons like today's plants.

Only one is cooled by light water, two are helium-cooled and the others have lead-bismuth, sodium or fluoride salt coolant. The latter three operate at low pressure, with significant safety advantage. The last has the uranium fuel dissolved in the circulating coolant. Temperatures range from 510°C to 1000°C, compared with less than 330°C for today's light water reactors, and this means that four of them can be used for thermochemical hydrogen production.

The sizes range from 150 to 1500 MWe (or equivalent thermal) , with the lead-cooled one optionally available as a 50-150 MWe "battery" with long core life (15-20 years without refuelling) as replaceable cassette or entire reactor module. This is designed for distributed generation or desalination.

At least four of the systems have significant operating experience already in most respects of their design, which provides a good basis for further R&D and is likely to mean that they can be in commercial operation well before 2030.

However, it is significant that to address non-proliferation concerns, the fast neutron reactors are not conventional fast breeders, ie they do not have a blanket assembly where plutonium-239 is produced. Instead, plutonium production takes place in the core, where burn-up is high and the proportion of plutonium isotopes other than Pu-239 remains high. In addition, new electrometallurgical reprocessing technologies will enable the fuel to be recycled without separating the plutonium.

In February 2005 five of the participants signed an agreement to take forward the R&D on the six technologies. The USA, Canada, France, Japan and UK agreed to undertake joint research and exchange technical information.

In January 2014 a new GIF Technology Roadmap Update was published. It confirmed the choice of the six systems and focused on the most relevant developments of them so as to define the R&D goals for the next decade. It suggested that the Gen IV technologies that are most likely to be deployed first are the sodium-cooled fast reactor, the lead-cooled fast reactor and the very high temperature reactor technologies. The molten salt reactor and the gas-cooled fast reactor were shown as furthest from demonstration phase. The third GIF symposium took place in Japan in May 2015 and considered progress with the six systems and in three methodology working groups.

MDEP

Closely related to GIF, but more focused on Generation III immediately, is the **Multinational Design Evaluation Programme** (MDEP) set up by the regulators. It was launched in 2006 by the US Nuclear Regulatory Commission (NRC) and the French Nuclear Safety Authority (ASN) to develop innovative approaches to leverage the resources and knowledge of national regulatory authorities reviewing new reactor designs. It involves the IAEA and 14 countries, and its secretariat is with the OECD Nuclear Energy Agency. Ultimately it aims to develop multinational regulatory standards for design of Generation IV reactors. The US Nuclear Regulatory Commission (NRC) has proposed a three-stage process culminating in international design certification for new reactor types, notably Generation IV types. Twelve countries are involved so far: Canada, China, Finland, France, India (from 2012), Japan, Korea, Russia, South Africa, Sweden (from 2013), UK, USA, and others which have or are likely to have firm commitments to building new nuclear plants may be admitted – the UAE is an associate member.

The MDEP pools the resources of its member nuclear regulatory authorities for the purpose of i) cooperating on safety reviews of designs of nuclear reactors that are under construction and undergoing licensing in several countries, and ii) exploring opportunities and potential for harmonization of regulatory requirements and practices. It also produces reports and guidance documents that are shared internationally beyond the MDEP membership. It has five design-specific working groups: EPR, AP1000, APR1400, VVER, ABWR, and three issue-specific ones: digital I&C; mechanical codes and standards; and vendor inspection cooperation.

In relation to Generation IV reactors, the NRC has called for countries involved in their development to develop common design requirements so that regulatory standards can be harmonized. The NRC has published its draft design requirements.

Meanwhile the MDEP is being used to share information among countries engaged in certifying particular new reactor designs, notably the EPR and AP1000, but with APR1400, VVER designs, and ABWR-ESBWR following. MDEP members are looking at different design codes to ensure that consistently high safety standards are achieved in different countries. Harmonization of design requirements will ultimately assist this, in the same way as has been achieved in civil aviation.

Associated ongoing programmes

While Russia was not initially part of GIF, one design corresponds with the BREST reactor being developed there, and Russia is now the main operator of the sodium-cooled fast reactor for electricity – another of the technologies put forward by the GIF.

India is also not involved with the GIF but is developing its own advanced technology to utilise thorium as a nuclear fuel. A three-stage program has the first stage well-established, with pressurised heavy water reactors (PHWR) fuelled by natural uranium to generate plutonium. Then fast breeder reactors (FBRs) use this plutonium-based fuel to breed U-233 from thorium, and finally advanced nuclear power systems will use the U-233. The spent fuel will be reprocessed to recover fissile materials for recycling. The two major options for the third stage, while continuing with the PHWR and FBR programmes, are an Advanced Heavy Water Reactor and subcritical Accelerator-Driven Systems.

A major project relevant to several Generation IV designs is investigating the use of actinide-laden fuel assemblies in fast reactors as part of the sodium-cooled fast reactor program. The Global Actinide Cycle International Demonstration (GACID) is being undertaken by France's atomic energy commission (CEA), Japan's Atomic Energy Agency (JAEA) and the US Department of Energy (DOE) under the US Advanced Fuel Cycle Initiative (AFCI). The first stage will lead to demonstration fuel containing minor actinides being used in Japan's Monju reactor.

GIF reactor technologies

There were originally six technologies chosen, but development on one has gone in two directions, so seven are listed in the Table below. Modification of the original designs is now under discussion due to the Fukushima accident, and in some cases that has extended the preparatory phase.

Gas-cooled fast reactors. Like other helium-cooled reactors which have operated or are under development, GFRs will be high-temperature units – 850°C. They employ similar reactor technology to the VHTR, suitable for power generation, thermochemical hydrogen production or other process heat. The reference GFR unit is 2400 MWt/1200 MWe, large enough for breakeven breeding, with thick steel reactor pressure vessel and three 800 MWt loops. For electricity, an indirect cycle with helium will be on the primary circuit, in the secondary circuit the helium gas will directly drive a gas turbine (Brayton cycle), and a steam cycle will comprise the tertiary circuit. It would have a self-generating (breeding) core with fast neutron spectrum and no fertile blanket. Robust nitride or carbide fuels would include depleted uranium and any other fissile or fertile materials as ceramic pins or plates, with plutonium content of 15 to 20%. As with the SFR, used fuel would be reprocessed on site and all the actinides recycled repeatedly to minimise production of long-lived radioactive wastes.

While General Atomics worked on the design in the 1970s (but not as fast reactor), none has so far been built. It is the only Gen IV design with no operating antecedent, so a prototype is not expected before 2022. However, a 75 MWt experimental technology demonstration GFR, Allegro, is planned by Euratom to be built from 2018. It will incorporate all the architecture and the main materials and components foreseen for the GFR without the power conversion system. Euratom, France, Japan and Switzerland have signed on to System Arrangements (SA) for the GFR under the Framework Agreement. See also European program section below.

An alternative GFR design has lower temperature (600-650ºC) helium cooling in a primary circuit and supercritical CO_2 at 550ºC and 20 MPa in a secondary system for power generation. This reduces the metallurgical and fuel challenges associated with very high temperatures. The General Atomics (GA) Energy Multiplier Module (EM^2) design is a 500 MWt, 240 MWe helium-cooled fast-neutron HTR operating at 850°C and fuelled with used PWR fuel or depleted uranium, plus some low-enriched uranium as starter. GA has teamed up with Chicago Bridge & Iron, Mitsubishi Heavy Industries, and Idaho National Laboratory to develop the EM^2, but it is not part of Gen IV program or mentioned in the 2014 roadmap.

Lead-cooled fast reactors. The LFR is a flexible fast neutron reactor which can use depleted uranium or thorium fuel matrices, and burn actinides from LWR fuel. Liquid metal (Pb or Pb-Bi eutectic) cooling is at atmospheric pressure by natural convection (at least for decay heat removal). Fuel is metal or nitride, with full actinide recycle from regional or central reprocessing plants. A wide range of unit sizes is envisaged, from factory-built "battery" with 15-20 year life for small grids or developing countries, to modular 300-400 MWe units and large single plants of 1400 MWe. Operating temperature of 550°C is readily achievable but 800°C is envisaged with advanced materials to provide lead corrosion resistance at high temperatures which would enable thermochemical hydrogen production. A two-stage development program leading to industrial deployment is envisaged: by 2025 for reactors operating with relatively low temperature and power density, and by 2040 for more advanced higher-temperature designs.

This corresponds with Russia's BREST fast reactor technology which is lead-cooled and builds on 80 reactor-years' experience of lead or lead-bismuth cooling, mostly in submarine reactors. However, these propulsion reactors were small, operated at low capacity factors, featured an epithermal (not fast) neutron spectrum and operated at significantly lower temperatures than those anticipated in Gen-IV LFRs. More immediately the GIF proposal appeared to arise from two experimental designs: the US STAR and Japan's LSPR, these being lead and lead-bismuth cooled respectively.

Initial development work on the LFR was focused on two pool-type reactors: SSTAR – Small Secure Transportable Autonomous Reactor of 20 MWe in USA; and the European Lead-cooled SYstem (ELSY) or European Lead Fast Reactor (ELFR) of 600 MWe in Europe.* In 2014 the leading developments were Russia's SVBR-100 and BREST-300, Europe's 300 MWt ALFRED, and Belgium's MYRRHA, with R&D focus on fuels and materials corrosion. In October 2015 Westinghouse announced that it had submitted an LFR project proposal for the DOE's upcoming investment in advanced reactor concepts demonstrable in the 2035 timeframe.

* SSTAR runs at 566°C and has integral steam generator inside the sealed unit, which would be installed below ground level. After a 20-year life without refuelling, the whole reactor unit would then be returned for recycling the fuel. The core is one metre high and 1.2 m diameter (20 MWe version). The ELSY or ELFR project is led

by Ansaldo Nucleare from Italy and was being financed by Euratom. The 600 MWe design was nearly complete in 2008 and a small-scale demonstration facility was planned, but little has been heard since then. It appears to be superseded by the ALFRED reactor. It runs on MOX fuel (with or without actinides) at 480°C and the molten lead is pumped to eight steam generators. Decay heat removal is by convection.

For the LFR, no System Arrangements (SA) have been signed, and collaborative R&D is pursued by interested members under the auspices of a provisional steering committee led by Japan and Euratom, joined by Russia in 2011. A technology pilot plant is envisaged in operation by 2021, followed by a prototype of a large unit and deployment of small transportable units. See also European program section below and Fast Reactor section in Russia Nuclear Power paper.

Molten salt reactors (now two variants): one a fast reactor with fissile material dissolved in the circulation fuel salt, and with solid particle fuel in graphite and the salt functioning only as coolant).

In a normal MSR, the uranium fuel is dissolved in the sodium fluoride salt coolant which circulates through graphite core channels to achieve some moderation and an epithermal neutron spectrum. The reference plant is up to 1000 MWe. Fission products are removed continuously and the actinides are fully recycled, while plutonium and other actinides can be added along with U-238, without the need for fuel fabrication. Coolant temperature is 700°C at very low pressure, with 800°C envisaged. A secondary coolant system is used for electricity generation, and thermochemical hydrogen production is also feasible.

Compared with solid-fuelled reactors, MSR systems have lower fissile inventories, no radiation damage constraint on fuel burn-up, no requirement to fabricate and handle solid fuel or solid used fuel, and a homogeneous isotopic composition of fuel in the reactor. These and other characteristics may enable MSRs to have unique capabilities and competitive economics for actinide burning and extending fuel resources.

During the 1960s the USA developed the molten salt fast reactor as the primary back-up option for the conventional fast breeder reactor, and a small prototype was operated for about four years. Recent work has focused on lithium and beryllium fluoride (FLiBe) coolant in a fast neutron spectrum (the MSFR) with dissolved thorium and U-233 fuel. MSFRs have large negative temperature and void coefficients. Other attractive features of the MSR fuel cycle include: the high-level waste comprising fission products only, hence shorter-lived radioactivity; small inventory of weapons-fissile material (Pu-242 being the dominant Pu isotope); low fuel use (the French self-breeding variant claims 50kg of thorium and 50kg U-238 per billion kWh); and safety due to passive cooling up to any size.

For the MSR, no system arrangements (SA) have been signed, and collaborative R&D is pursued by interested members under the auspices of a provisional steering committee involving France, Russia and Euratom. There will be a long lead time to prototypes, and the R&D orientation has changed since the project was set up, due to increased interest. It now has two baseline concepts:

- The Molten Salt Fast Neutron Reactor (MSFR), which will take in thorium fuel cycle, recycling of actinides, closed Th/U fuel cycle with no U enrichment, with enhanced safety and minimal wastes.
- The Advanced High-Temperature Reactor (AHTR) – also known as the fluoride salt-cooled high-temperature reactor (FHR) – with the same graphite and solid fuel core structures as the VHTR and molten salt as coolant instead of helium, enabling power densities 4 to 6 times greater than HTRs and power levels up to 4000 MWt with passive safety systems. The TMSR Research Centre has small solid-fuel prototype under construction at Shanghai Institute of Nuclear Applied Physics (SINAP, under the China Academy of Sciences) which had a 2015 target for operation.

The GIF 2014 Roadmap said that a lot of work needed to be done on salts before demonstration reactors were operational, and suggested 2025 as the end of the viability R&D phase.

Sodium-cooled fast reactors. The SFR uses liquid sodium as the reactor coolant, allowing high power density with low coolant volume, at low pressure. It builds on some 390 reactor-years experienced with sodium-cooled fast neutron reactors over five decades and in eight countries, and was initially the main technology of interest in GIF, and remains at the forefront despite needing a sealed coolant system. A variety of fuels is possible. Most SFR plants so far have had a core plus blanket configuration, but new designs are likely to have all the neutron action in the core. Other R&D is focused on safety in loss of coolant scenarios, and improved fuel handling.

The SFR utilises depleted uranium as the fuel matrix and has a coolant temperature of 500-550°C enabling electricity generation via a secondary sodium circuit, the primary one being at near atmospheric pressure. Three variants are proposed: a 50-150 MWe modular type with actinides incorporated into a U-Pu metal fuel requiring electrometallurgical processing (pyroprocessing) integrated on site; a 300-1500 MWe pool-type version of this; and a 600-1500 MWe loop-type with conventional MOX fuel, poterntially with minor actinides, and advanced aqueous reprocessing in central facilities elsewhere.

Early in 2008, the USA, France and Japan signed an agreement to expand their cooperation on the development of sodium-cooled fast reactor technology. The agreement relates to their collaboration in the International Framework for Nuclear Energy Cooperation (IFNEC), (formerly Global Nuclear Energy Partnership), aimed at closing the nuclear fuel cycle through the use of advanced reprocessing and fast reactor technologies, and seeks to avoid duplication of effort.

Two significant large SFRs are starting up: the BN-800 at Beloyarsk in Russia (started in 2014 but not grid-connected by mid-2015) and the Kalpakkam PFBR of 500 MWe in India (expected in late 2015). The BN-800 is largely an experimental reactor. GIF observes that the technology is 'deployable in the very near-term for actinide management.' Much of the ongoing R&D focus will be on fuels.

Euratom, China, France, Japan, Korea and the USA have signed on to System Arrangements (SA) for the SFR under the Framework Agreement, and in 2011 Russia joined them. Several Project Arrangements are within the SFR system: the Safety and Operation PA; the Advanced Fuel PA; the Global Actinide Cycle International Demonstration (GACID) PA; the Component Design and Balance-Of-Plant PA; and the System Integration and Assessment PA. See also European program section below.

Supercritical water-cooled reactors (SCWR). This is a very high-pressure water-cooled reactor which operates above the thermodynamic critical point of water (374ºC, 22 MPa) to give a thermal efficiency about one third higher than today's light water reactors from which the design evolves. The supercritical water (25 MPa and 510-550°C) directly drives the turbine, without any secondary steam system,* simplifying the plant. Two design options are considered: pressure vessel and pressure tube. Passive safety features are similar to those of simplified boiling water reactors. Fuel is uranium oxide, enriched in the case of the open fuel cycle option. The core may use thermal neutron spectrum with light or heavy water moderation, or be a fast reactor with full actinide recycle based on conventional reprocessing. Since the SCWR builds both on much BWR experience and that from hundreds of fossil-fired power plants operated with supercritical water, it can readily be developed, and the operation of a 30 to 150 MWe technology demonstration reactor is targeted for 2022.

Japanese studies on a pressure-vessel design have confirmed target thermal efficiency of 44% with 500°C core outlet temperature, and estimate a potential cost reduction of 30% compared with present PWRs. Safety features re expected to be similar to ABWRs. Canada is developing a pressure-tube design with heavy water moderation.

Euratom, Canada and Japan have signed on to System Arrangements (SA) for the SCWR under the Framework Agreement, in 2011 Russia joined them, followed by China in 2014. Project arrangements are pending for thermal-hydraulics and safety. Pre-conceptual SCWR designs include Candu (Canada), LWR (Euratom) and Fast Neutron (Japan).

* Today's supercritical coal-fired plants use supercritical water around 25 MPa which have "steam" temperatures of 500 to 600ºC and can give 45% thermal efficiency. At ultra supercritical levels (30+ MPa), 50% thermal efficiency may be attained. Over 400 such plants are operating world-wide.

Supercritical fluids are those above the thermodynamic critical point, defined as the highest temperature and pressure at which gas and liquid phases can co-exist in equilibrium. They have properties between those of gas and liquid. For water the critical point is at 374°C and 22 MPa, giving it a "steam" density one third that of the liquid so that it can drive a turbine in a similar way to normal steam.

Very high-temperature gas reactors. These are graphite-moderated, helium-cooled reactors, based on substantial experience.Euratom, Canada and Japan have signed on to System Arrangements (SA) for the SCWR under the Framework Agreement. Project Arrangements are pending for thermal-hydraulics and safety. Pre-conceptual SCWR designs include Candu (Canada), LWR (Euratom) and Fast Neutron (Japan).

The core can be built of prismatic blocks such as the Japanese HTTR and General Atomics' earlier GTMHR design and others in Russia, or it may be pebble bed such as the Chinese HTR-10 or HTR-PM and the PBMR formerly under development in South Africa. Outlet temperature of over 900°C and aiming for 1000ºC enables thermochemical hydrogen production via an intermediate heat exchanger, with electricity cogeneration, or direct high-efficiency driving of a gas turbine (Brayton cycle). At lower outlet temperatures, the Rankine steam cycle may be used for electricity generation, and this is the focus for demonstration projects. Modules of 600 MW thermal are envisaged.

There is some flexibility in fuels, but no recycle initially. Fuel is in the form of TRISO (tristructural-isotropic) particles less than a millimetre in diameter. Each has a kernel (ca. 0.5 mm) of uranium oxycarbide (or uranium dioxide), with the uranium enriched up to 20% U-235, though normally less. This is surrounded by layers of carbon and silicon carbide, giving a containment for fission products which is stable to over 1600°C. The particles may be incorporated in billiard ball sized pebbles, or in prismatic graphite blocks. The VHTR has potential for high burn-up (150-200 GWd/t), completely passive safety, low operation and maintenance costs, and modular construction.

The 2014 GIF Roadmap says that a 600 MWt VHTR dedicated to hydrogen production could yield over two million normal cubic metres per day. An R&D priority is qualification of TRISO fuel for operation up to 1250°C and 200 GWd/t burn-up, though US development has attained this, and also its robustness for hundreds of hours at 1,600, 1,700 and 1,800°C. However, in the short term, electricity production and industrial processes based on high temperature steam that require outlet temperatures of (700-850°C) hold most potential.

One question to be resolved regards used fuel. Some synergy with used LWR fuel is possible, and the graphite may be recycled, but much remains to be worked out.

Euratom, France, Japan, China, Korea, Switzerland and the USA originally signed on to the System Arrangement (SA) for the VHTR under the Framework Agreement. Two Project Arrangements were signed within the VHTR system: the Fuel and Fuel Cycle PA and the Hydrogen Production PA.

The HTR-PM demonstration unit in China is under construction at Shidaowan, and will pave the way for a commercial version of the VHTR.

Generation IV reactor designs under development by GIF

	neutron spectrum (fast/ thermal)	coolant	temperature (°C)	pressure*	fuel	fuel cycle	size(s) (MWe)	uses
Gas-cooled fast reactors	fast	helium	850	high	U-238 +	closed, on site	1200	electricity & hydrogen
Lead-cooled fast reactors	fast	lead or Pb-Bi	480-570	low	U-238 +	closed, regional	20-180** 300-1200 600-1000	electricity & hydrogen
Molten salt fast reactors	fast	fluoride salts	700-800	low	UF in salt	closed	1000	electricity & hydrogen
Molten salt reactor - **Advanced High-temperature reactors**	thermal	fluoride salts	750-1000		UO_2 particles in prism	open	1000 - 1500	hydrogen

	neutron spectrum (fast/thermal)	coolant	temperature (°C)	pressure*	fuel	fuel cycle	size(s) (MWe)	uses
Sodium-cooled fast reactors	fast	sodium	500-550	low	U-238 & MOX	closed	50-150 600-1500	electricity
Supercritical water-cooled reactors	thermal or fast	water	510-625	very high	UO_2	open (thermal) closed (fast)	300-700 1000-1500	electricity
Very high temperature gas reactors	thermal	helium	900-1000	high	UO_2 prism or pebbles	open	250-300	hydrogen & electricity

* high = 7-15 MPa
\+ = with some U-235 or Pu-239
** 'battery' model with long cassette core life (15-20 yr) or replaceable reactor module.

European program from 2010

The European Commission in 2010 launched the European Sustainable Nuclear Industrial Initiative (ESNII), which will support three Generation IV fast reactor projects as part of the EU's plan to promote low-carbon energy technologies. Other initiatives supporting biomass, wind, solar, electricity grids and carbon sequestration are in parallel. ESNII will take forward: the Astrid sodium-cooled fast reactor (SFR) proposed by France for construction there, the Allegro gas-cooled fast reactor (GFR) supported by central and eastern Europe, and the ALFRED lead-cooled fast reactor

(LFR) technology pilot in Romania, supported by the MYRRHA lead-bismuth facility in Belgium.

The aim of ESNII is to demonstrate Gen IV reactor technologies that can close the nuclear fuel cycle, provide long-term waste management solutions, and expand the applications of nuclear fission beyond electricity production to hydrogen production, industrial heat and desalination. ESNII is designed to combine European capabilities in fast neutron reactor R&D with industrial capability to build the prototypes and develop supporting infrastructure.

The total estimated cost to ESNII of deploying these Gen IV prototypes past 2020 was €10.8 billion: €5 billion for Astrid, €1.96 billion is for ALFRED and MYRRHA, a technology pilot and a later LFR demonstrator, and €1.2 billion for Allegro. Supporting infrastructure is projected to cost €2.65 billion. The 2010-12 ESNII budget was €527 million, including €329 million for Astrid.

Astrid SFR is led by the French CEA, involves EdF and Areva, and is supported by a French government loan of €651 million. Astrid is based on about 45 reactor-years of operational experience in France and will be rated 400 to 600 MWe. It is expected to be built at Marcoule from 2020, with the unit being connected to the grid in 2025.

Allegro GFR is to be built in eastern Europe, and is more innovative. It is rated at 75 MWt and is being developed out of the EU GoFastR project. The ALLIANCE project (Preparation of Allegro – Implementing Advanced Nuclear Fuel Cycle in Central Europe) was then launched in 2012 to continue the elaboration of basic documents needed for high-level decisions and licencing of Allegro. The main nuclear parameters (power density, burn-up *etc.*) would be similar to those of the planned 2400 MWth GFR. The core built up from the initial fuel type (MOX) will be replaced by a core of ceramic fuel for the second half of Allegro operation. The Czech Republic, Hungary and Slovakia are making a joint proposal to host the project, with French CEA support. Allegro is expected to begin construction in 2020 and operate from 2025. The industrial demonstrator would follow it.

In mid-2013 four nuclear research institutes and engineering companies from central Europe's Visegrád Group of Nations (V4) agreed to establish a centre for joint research, development and innovation in Generation IV nuclear reactors. The V4G4 Centre of Excellence is being set up by scientific and research engineering company ÚJV Řež AS of the Czech Republic, the Academy of Sciences Centre for Energy Research of Hungary, Poland's National Centre for Nuclear Research, and engineering company VUJE AS of Slovakia. It is focused on gas-cooled fast reactors such as Allegro.

ALFRED LFR technology demonstrator – the Advanced Lead Fast Reactor European Demonstrator – of about 300 MWt is seen as a prelude to an industrial demonstration unit of about 300-400 MWe. ALFRED will employ mixed oxide (MOX) fuel, with about 17% plutonium in equilibrium, and able to recycle minor actinides as about 1% of feed. Construction of ALFRED could begin in 2020 and the unit could start operating after 2025.

A consortium was set up in December 2013 for ALFRED's construction, comprising Italy's National Agency for New Technologies, Energy and the Environment (ENEA), Ansaldo Nucleare, and Romania's Nuclear Research Institute (Institutul de Cercetari Nucleare, ICN). The group is to be known as the Fostering Alfred Construction (Falcon) consortium, which will be expanded through the participation of further European organizations. The total cost of the project is put at some €1.0 billion. ALFRED will be built at ICN's facility in Mioveni, near Pitesti in southern Romania, where a fuel manufacturing plant is in operation for the country's two operating Candu reactors.

The **MYRRHA LFR** project is initially a 57 MWt accelerator-driven system with a liquid lead-bismuth (Pb-Bi) spallation target that in turn couples to a Pb-Bi cooled, subcritical fast nuclear core. Later it will become a European fast neutron technology pilot plant for lead and a multi-purpose research reactor. Belgium's SCK-CEN is leading the project and will provide a total of about €450 million. The unit is rated at 100 thermal MW and has entered a new phase in the validation process, with the second phase of the front-end engineering and design to begin around late 2015, with operation expected in 2020. It will be built at SCK-CEN's Mol site. A reduced-power model of MYRRHA called Guinevere started up at Mol in March 2010.

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