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SUMMARY

The main purpose of this paper is to present the current status of development of the Lead-cooled Fast Reactor (LFR) in Generation IV (GEN IV), including the European contribution, to identify needed R&D and to present the corresponding GEN IV International Forum (GIF) R&D plan [1] to support the future development and deployment of lead-cooled fast reactors.

The approach of the GIF plan is to consider the research priorities of each member country in proposing an integrated, coordinated R&D program to achieve common objectives, while avoiding duplication of effort. The integrated plan recognizes two principal technology tracks:

- a small, transportable system of 10–100 MWe size that features a very long refuelling interval, and
- a larger-sized system rated at about 600 MWe, intended for central station power generation.

This paper provides some details of the important European contributions to the development of the LFR. Sixteen European organizations have, in fact, taken the initiative to present to the European Commission the proposal for a Specific Targeted Research and Training Project (STREP) devoted to the development of a European Lead-cooled System, known as the ELSY project; two additional organizations from the US and Korea have joined the project. Consequently, ELSY will constitute the reference system for the large lead-cooled reactor of GEN IV. The ELSY project aims to demonstrate the feasibility of designing a competitive and safe fast power reactor based on simple technical engineered features that achieves all of the GEN IV goals and gives assurance of investment protection.

As far as new technology development is concerned, only a limited amount of R&D will be conducted in the initial phase of the ELSY project since the first priority is to define the design guidelines before launching a larger and expensive specific R&D program. In addition, the ELSY project is expected to benefit greatly from ongoing lead and lead-alloy technology development already being carried out in different institutes participating in this STREP. This is particularly true in Europe where a large R&D program associated with the development of Accelerator Driven Systems (ADS) is being actively pursued.

The general objective of the ELSY project is to design an innovative lead-cooled fast reactor complemented by an analytical effort to assess the existing knowledge base in the field of lead-alloy coolants (i.e., lead-bismuth eutectic (LBE) and also lead/lithium) in order to extrapolate this knowledge base to pure lead. This analysis effort will be complemented with some limited R&D activities to acquire missing or confirmatory information about fundamental topics for ELSY that are not sufficiently covered in the ongoing European ADS program or elsewhere.

Considering the significant commonality of R&D that can be found between the small, transportable system, and the medium-or large-sized system of the two GEN IV tracks, the GIF plan proposes coordinated research with a single demonstration facility that can serve the

R&D needs of both approaches while reducing the unnecessary expense of duplicate major test facilities.

List of Acronyms:

ELSY	European Lead-cooled SYstem	LFR	Lead-cooled Fast Reactor
ADS	Accelerator Driven System	LWR	Light Water Reactor
DBC	Design Basis Conditions	MA	Minor Actinides
DEC	Design Extended Conditions	MOX	Mixed OXide fuel
DEMETRA	DEvelopment and assessment of structural materials and heavy liquid MEtal technologies for TRAnsmutation systems),	MS&FW	Main Steam and Feed Water system
DHR	Decay Heat Removal	R&D	Research and Development
FP6	Euratom Sixth Framework Programme	SC	Steering Committee
GEN IV	Generation IV	SFR	Sodium Fast Reactor
GIF	Generation IV International Forum	STREP	Specific Targeted Research and Training Projects
IP-EUTOTRANS	EUROpean Research Programme for the TRANSmutation of High Level Nuclear Waste in an Accelerator Driven System	VELLA	Virtual European Lead Laboratory
LBE	Lead-Bismuth Eutectic		

A. LFR IN GENERATION IV

The Generation IV (GEN IV) Technology Roadmap [2], prepared by GIF member countries, identified the six most promising advanced reactor systems and fuel cycle concepts and the R&D necessary to advance these concepts for potential deployment.

Among the promising reactor technologies being considered by the GIF, the LFR has been identified as a technology with great potential to meet the needs for both remote sites and central power stations.

In the GEN IV technology evaluations, the LFR system was top-ranked in sustainability because a closed fuel cycle is considered, and in proliferation resistance and physical protection because it employs a long-life core. It was rated good in safety and economics. The safety was considered to be enhanced by the choice of a relatively inert coolant. 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 to be deployable by 2025.

The LFR system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. The LFR can also be used as a burner of all actinides from spent fuel by using inert matrix fuel and as a burner / breeder with thorium matrices.

A Lead-cooled Fast Reactor Provisional R&D Steering Committee was established in the year 2005 under the auspices of the GIF initiative with the following members: Craig F.

Smith (USA), Mamoru Konomura (Japan), Kune Y. Suh (South Korea), Luciano Cinotti (Euratom).

The Provisional R&D Steering Committee has prepared a DRAFT OF THE RESEARCH & DEVELOPMENT PLAN FOR THE LEAD-COOLED FAST REACTOR (LFR) [1] with molten lead as the reference coolant option, lead-bismuth as backup coolant, and a dual-track design approach.

This approach consists of the design of a small transportable system of 10–100 MWe size that features a very long refuelling interval, and of a larger system, rated at about 600 MWe, intended for central station power generation.

A.1 The small transportable system

Two key technical aspects of the envisioned small LFR are the use of lead (Pb) as coolant and a long-life sealed or cartridge-core architecture in a small, modular system intended for deployment with small grids or remote locations.

The small LFR envisioned is the Small Secure Transportable Autonomous Reactor (SSTAR) concept, which is a small, modular, fast reactor. The main mission of the 20 MWe (45 MWth) SSTAR is to provide incremental energy generation to match the needs of developing nations and remote communities lacking electrical grid connections, such as those that exist in Alaska or Hawaii, island nations of the Pacific Basin, and elsewhere. This may be a niche market where energy production costs that are higher than those of a large-scale nuclear power plants can be accepted.

Design features of the reference SSTAR include a 20-to-30-year-lifetime sealed core, a natural circulation primary coolant system, autonomous load following without control rod motion, and use of a supercritical CO2 (S-CO_2) energy conversion cycle. The incorporation of inherent thermo-structural feedbacks imparts walk-away passive safety, while the use of a sealed cartridge core with a 20-year or longer cycle time between refuelling imparts strong proliferation resistance.

The challenging aspect of the core design is to establish the necessary features of a 20 to 30-year-life core and to determine core parameters that impact feedback coefficients.

System thermal hydraulic studies are essential to establish the parameters for potential natural circulation cooling in the primary system, to identify any safety issues to be addressed in subsequent design, and to establish parameters for ensuring passive safety response. Passive safety response can be designed into the reactor core and plant based on current experience and passive safety design principles. However, the magnitudes of feedback coefficients for a given design and integral behavior of a reactor plant must be verified through further analysis and experiments.

Experience with LWRs and previous fast reactor plants and concepts indicates that large containments, needed to contain gaseous reaction and fission products, drove such plants to large economies of scale. This must be avoided if the small LFR is to be financially viable. Therefore, the driving factors for the SSTAR containment design must be evaluated bearing in mind that large-size containment should be avoided.

Figure 1 provides a sketch of the currently envisioned SSTAR small LFR system concept and operating parameters.

Preliminary designs of larger plants cooled by LBE have already been carried out in Korea (PEACER-300 and PEACER-550) and in Japan (with a power ranging up to 750 MWe).

It is a natural development to select the use of pure lead as a coolant since it is chemically inert in comparison to sodium and is less corrosive, of lesser radiological concern when activated and cheaper than LBE. Lead has good neutronic characteristics that are unique among the coolants for fast reactors.

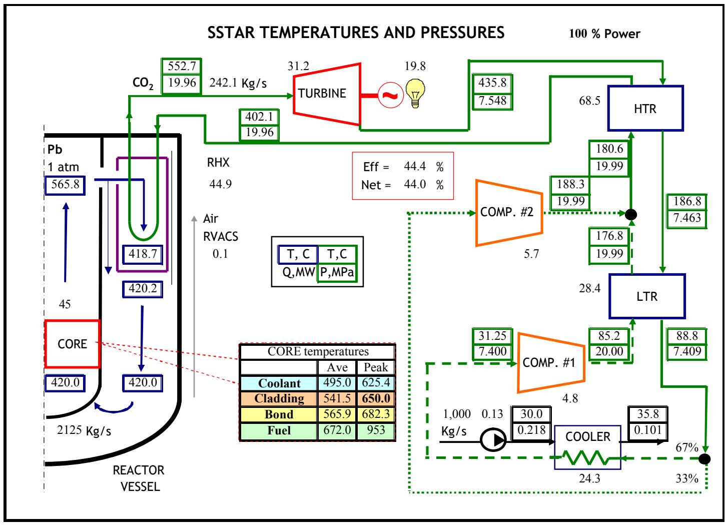

Figure 1- SSTAR Preliminary Design Concept and Operating Parameters

A.2 The large-sized system for central station power generation

In the third call of the FP6, Sixteen European organizations joined together to take the initiative to present to the European Commission the proposal for a Specific Targeted Research and Training Project (STREP) devoted to the development of a European Lead-cooled System (ELSY). Two additional organizations from the US and Korea have joined the project, and it is expected that other European and non European Partners may join to design and develop a common large-scale LFR (see Table 1).

The ELSY project aims at demonstrating that it is possible to design a competitive and safe Lead-cooled fast power reactor using simple engineered features. This prospective is appealing also for private investors who have offered to participate in the initiative. This would create the conditions for advancing the ELSY activity beyond FP6.

The use of compact, in-vessel steam generators and a simple primary circuit with possibly all internals being removable are among the reactor features needed for competitive electric energy generation and long-term protection of investment.

The tentative parameters of ELSY are specified in Table 2 and the main milestones of the project are presented in Table 3.

<u>Plant power</u>

The power plant is tentatively sized at 600 MWe because only plants of the order of several hundreds MWe will be economically productive on the existing, well-interconnected grids of Europe. On the other hand, a plant larger than the reference size will require an increase in the lead mass and the associated mechanical loads on the reactor vessel and its supporting structure.

Coolant

A large experience base exists on LBE in Russia and elsewhere in the world because of evaluation of its use in subcritical reactors which are low-power and without significant requirements for electricity production.

Table 1- Organizations involved in the ELSY project

Partic. no.	Participant organisation	Participant short name	Country
1	Ansaldo Energia S.p.A, Nuclear Division	ANSALDO	Italy
2	AGH, Akademia Górniczo-Hutnicza	AGH	Poland
3	Centro Elettrotecnico Sperimentale Italiano	CESI	Italy
4	Inter Universities Consortium for Nuclear Technological Research	CIRTEN	Italy
5	Centre National de la Recherche Scientifique	CNRS	France
6	Empresarios Agrupados Internacional S.A.	EA	Spain
7	Electricité de France	EDF	France
8	Ente Per Le Nuove Tecnologie, L'energia e L'ambiente	ENEA	Italy
9	Forschungszentrum Karlsruhe GmbH	FZK	Germany
10	Institute for Nuclear Research	INR	Romania
11	European Commission, Joint Research Centre	JRC	Europe
12	Royal Institute of Technology-Stockholm	KTH	Sweden
13	Nuclear Research and Consultancy Group	NRG	Netherlands
14	Ustav jaderneho vyzkumu Rez, a.s. (Nuclear Research Institute Rez, plc.)	UJV	Czech Republic
15	Paul Scherrer Institut	PSI	Switzerland
16	Studiecentrum voor Kernenergie•Centre d'Etude de l'énergie Nucléaire	SCK•CEN	Belgium
17	Lawrence Livermore National Laboratory	LLNL	USA
18	Seoul National University, Nuclear Engineering Department(NED), Nuclear Transmutation Energy Research Center of Korea (NUTRECK)	SNU	Korea

Since lead is much more abundant (and less expensive) and hence more available in comparison with bismuth, in case of deployment of a large number of reactors, the selection of pure lead as a coolant offers enhanced sustainability.

The generation of highly radioactive, and hence decay-heat generating polonium as a coolant activation product is much lower than in the case of LBE. The omission of bismuth in the coolant, therefore, reduces the problems associated with decay heat removal.

Operation at a higher temperature, required by the use of pure lead, would generally be necessary also in the case of LBE to improve plant efficiency and to avoid excessive embrittlement of structural material subjected to fast neutron flux at low-temperature. The risk of lead freezing is reduced by the choice of the pool-type reactor.

Coolant circulation

The choice of a large reactor power suggests the use of forced circulation to shorten the reactor vessel, thereby avoiding excessive coolant mass and alleviating mechanical loads on the reactor vessel.

It is known that, thanks to the favorable neutronic characteristics of lead as a coolant, the fuel rods of a lead-cooled reactor, similarly to LWRs, can be spaced further apart than in the case of sodium as a coolant, and this results in a low pressure drop across the core. The needed pump head, in spite of the higher density of lead, can therefore be kept low (of the order of one to two bars) with a reduced requirement for pumping power.

As in the European 80 MW LBE-cooled XADS [3], a simple gas lift as pumping system with 24 parallel riser pipes could be selected, instead of mechanical pumps, to enhance the primary coolant natural circulation to the specified flow rate. A test section of this gas lift system has been installed in the CIRCE facility (at the ENEA site of Brasimone) with one full scale riser pipe. The test results confirm the suitability of gas lift for a small-power reactor, but also show decreasing efficiency at higher flow rates, a fact that makes its applicability questionable for a large plant such as ELSY. Therefore, for ELSY, it is expected that the use of mechanical pumps will be more suitable.

Table 2 - Tentative parameters of the ELSY plant

PLANT CHARACTERISTIC	TENTATIVE PLANT PARAMETERS
Power	600MWe
Thermal efficiency	40 %
Primary coolant	Pure lead
Primary system	Pool type, compact
Primary coolant circulation (at power)	Forced
Primary coolant circulation for DHR	Natural circulation + Pony motors
Core inlet temperature	~ 400°C
Core outlet temperature	~ 480°C
Fuel	MOX with consideration also of nitrides and dispersed minor actinides
Fuel handling	ELSY will seek innovative solutions
Main vessel	Austenitic stainless steel, hanging, short-height
Safety Vessel	Anchored to the reactor pit
Steam Generators	Integrated in the main vessel
Secondary cycle	Water-supercritical steam
Primary Pumps	Mechanical, in the hot collector
Internals	Removable to the greatest possible extent, (objective: all removable)
Inner Vessel	Cylindrical
Hot collector	Small-volume, above the core
Cold collector	Annular, outside the Inner Vessel, free level higher than free level of hot collector
DHR coolers	Immersed in the cold collector
Seismic design	2D isolators supporting the main vessel

Decay heat removal

According to the predicted low primary system pressure loss and the favorable thermodynamic characteristics of lead, decay heat can be removed in natural circulation.

Core thermal cycle

The proposed thermal cycle includes a 400 °C core inlet temperature to provide sufficient margin from the melting point of lead, and a relatively low 480 °C core outlet temperature to benefit from advantages in term of reduced corrosion, improvement of mechanical characteristics (reduced creep) of the structural steels, and reduced thermal shocks in transient conditions.

In terms of efficiency of electricity energy production, the elimination of the secondary coolant system would compensate for the effects of the reduction of the core outlet temperature (by 62 K in comparison to SuperPhenix Na cooled reactor (SPX1)). In fact, a supercritical cycle becomes possible, even with a lower steam temperature.

Table 3 - Main milestones of the ELSY project

First year	Second year	Third year
- General Specifications - Preliminary core configuration - Main component configuration - Lead data base	- Core design (with wrappers) - Core design (open square fuel assembly) - Vessel design - SG design - Pump design - DHR design - Fuel handling design - MS&FW definition - DBC accident analysis	- Cost estimate - Compliance with the GEN IV goals - Definition of future R&D needs - Reference core selection - Impact of loading MA - Impact of nitride fuel - Functional and mechanical sizing of main components - 3D modelling of the reactor - Digital process simulation and information System - Reference plant layout - DEC accident analysis - Construction material for pumps - Qualification of coatings and/or surface treatments

In Figure 2, the anticipated thermal cycle is compared with the technological limits extrapolated from the European experience on LBE. Prospective R&D activities will be conducted to confirm that lead is less aggressive than LBE towards structural materials. [4] The reactor vessel, designed to operate at the cold temperature of 400°C, would be in a safe condition even assuming that the oxygen control in the melt be temporarily lost [5]. All reactor internals will have to operate in a temperature regime where it is necessary to rely on oxygen control, whereas fuel cladding could be surface-treated (aluminization seems to be a promising route) for a greater safety margin. Increasing the core outlet temperature to about 550°C, as for the case of the SFR, would create an unjustified technological risk without any guarantee of technological success in the timeframe indicated by GIF. An improved thermal cycle at higher temperature could be adopted in the longer term, as new materials will be made available.

Seismic design

The reactor vessel will be shortened as much as possible to reduce the seismic loads; 2D seismic supports will be provided to minimize the effect of the high lead density.

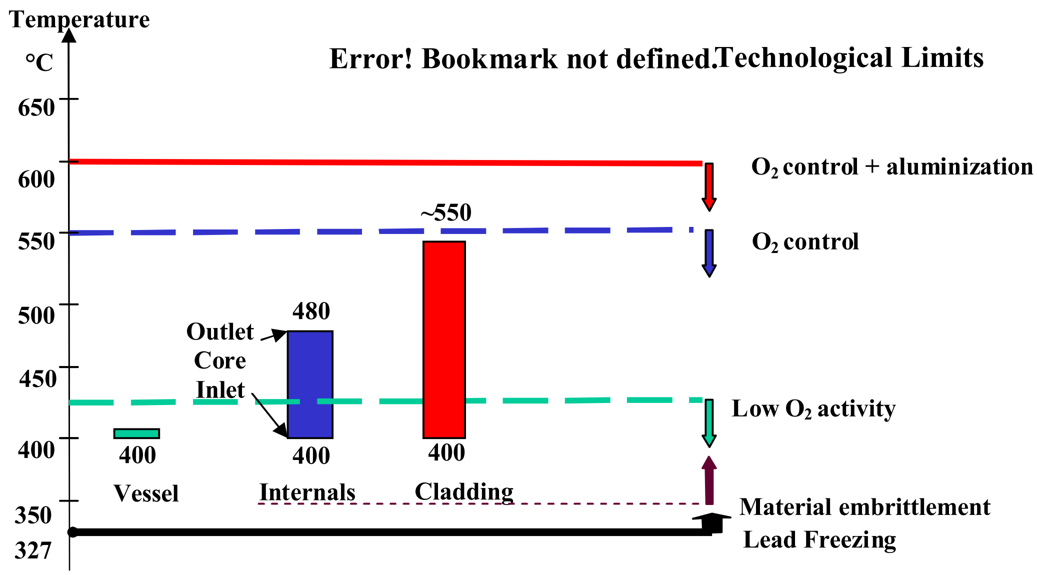

Figure 2 - Proposed ELSY thermal cycle

Primary System

The closed primary coolant loop in the pool-type reactor suggests the hydraulic separation of hot and cold pools by means of an inner vessel. A classical design for the inner vessel is a simple or double-walled membrane (Redan in French) spanning the reactor vessel. The shape of the Redan is complicated by the penetrations of the intermediate heat exchangers (IHX) and primary pumps. Besides the structural complication, this kind of pool separation has the inherent disadvantages of reduced volume of the cold pool and the immersion of DHR coolers in the hot pool, a configuration that may hinder the prompt onset of natural circulation in case of loss of station service power. A different shape of inner vessel, the cylindrical inner vessel, is the backup design proposed in a few fast reactors to overcome the disadvantages of the Redan. None of the proposed backup designs, however, is fully satisfactory, particularly regarding the often bulky solutions adopted to connect the component parts of the primary loop.

An example of an improved scheme to be evaluated as a starting point for the primary system of ELSY is the cylindrical inner vessel concept represented in Figure 3.

The steam generator (SG) and primary pump (PP) assembly made of two SGUs and one PP arranged between the SGUs, and casing, is an integral part of the primary loop, i.e. from PP suction to SGU exit. It is of kidney-shaped cross section because it is arranged in the annular space between cylindrical inner vessel and reactor vessel and hence immersed in the cold pool. It is supported by, and hangs freely from, the reactor vessel lid. The only connection with the reactor internals is by the suction pipe of the PP that is engaged in the piston seal at the upper end of the elbow welded to the inner vessel. Thus, the whole assembly can be flasked in and out of the reactor vessel. Hot lead is pumped into the pool above the PP and SGU and driven shell-side downwards across the SGU helical-tube bundle into the cold pool. The free levels of the coolant depend on the pressure drop relationships among the different sections of the primary loop and, at normal steady-state operation, the free level of the hot pool inside the casing is higher than the free level of the cold pool outside that is higher, in turn, than the free level of the hot pool above the core enclosed by the inner vessel. In case of loss of forced circulation, the available head would ensure the prompt onset of the natural circulation, and hence preserve the safety function of core cooling from the very beginning of the transient.

The ELSY design activity that is proposed to be included also in the GIF plan consists of:

- Definition of the design objectives, costs, future R&D needs and approaches to achieve compliance with the GEN IV goals.
- Core design and performance assessment.
- Main components and systems design.
- System integration.
- Safety and transient analysis.

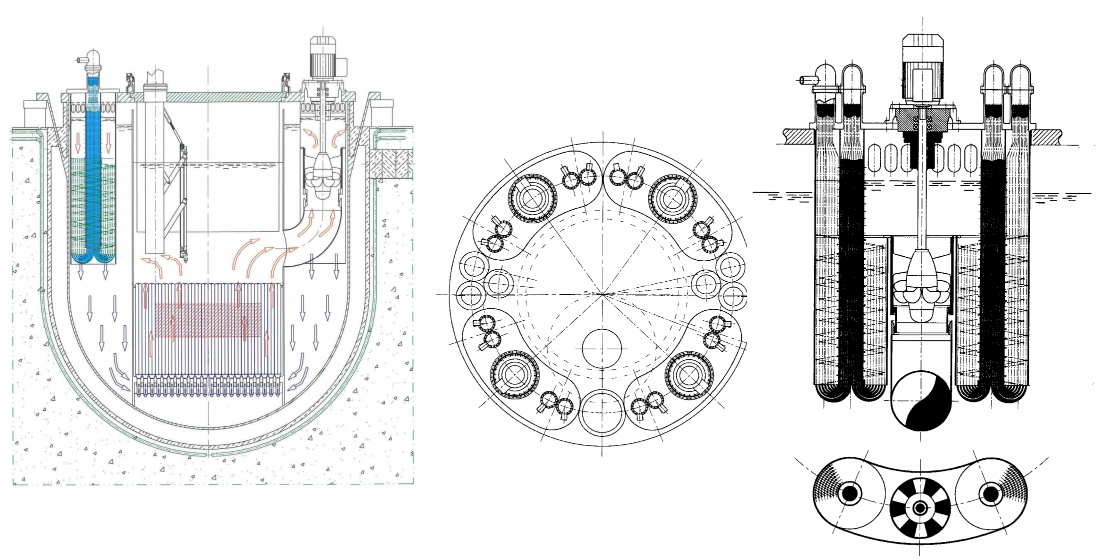

Figure 3 - Preliminary scheme of the ELSY Reactor with, at the right, the sectional views of one of the four identical assemblies, each made of two steam generating units and one primary pump.

Design objectives, cost estimate, future R&D needs and compliance with the GEN IV goals

This activity will start with the definition of the design objectives, particularly referring to GEN IV goals and the definition of the ELSY requirements and general specifications, based on the requirements of the Utilities, whenever applicable.

Design reviews will be held periodically in order to verify the compliance of the ongoing project with the project rules. At the end of the design activity, support R&D needs will be listed, a study cost estimate will be carried out and the compliance report of the final project with the GEN IV goals will be issued.

Core design and performance assessment

This activity consists of the general core specification and subsequent identification of the more promising core configurations, including the impact of the presence of MA on the performance of the reference MOX core, in order to assess the system capability to burn its own MA and, not least, to evaluate the impact of different amounts of MAs on safety and control capability of the reactor. Furthermore, the overall impact of a nitride fuel, including core performance, will be assessed in the frame of the same core configuration. The Russian choices embodied in the BREST [6] core, particularly the wrapperless fuel elements and the heterogeneous-diameter fuel rods, will be duly considered as potential design options.

Main components and systems design

This activity consists of the preliminary design of the following main components, whereby attention will be particularly given to the choice of the structural steels in contact with molten lead:

- the Reactor Vessel, with focus on the seismic analysis of its support system;
- the Steam Generator Unit, with focus on in-service inspection (ISI) of its pressure boundary;
- the Primary Pump, with focus on the choice of innovative construction material suitable for use in the corrosive/erosive molten lead environment and on low pressure drop across the pump with coolant in natural circulation pattern;
- the Decay Heat Removal System, with focus on its passive behavior and reliability, thanks to the criterion of combined simplicity, redundancy and diversity;
- the selection of the steam and feed-water system cycle, with focus on the system efficiency.

System integration

This activity consists of the conceptual design/optimization of the system reference configuration, including back-up options, the containment system and the overall plant layout. Attention will be particularly given to the reactor configuration and the interfaces among core, SGUs, primary pumps, DHR coolers, and refuelling System. The assessment of the effects of sloshing lead during an earthquake will be made as well as of pressure wave damping as a consequence of an accidental SG tube break. An additional challenging activity will be the analysis of the possibility of replacement of in-vessel components in case it becomes necessary during the plant lifetime.

Safety and transient analysis

This activity consists of:

- Evaluation of the operational plant behavior and the plant modelling as input/surveillance of the ongoing project activity;
- Identification of representative accident initiators within Design Basis Conditions (DBC) and Design Extension Conditions (DEC);
- Safety analyses of plant accidents representative of DBC;
- Investigation of accidents within (DEC) including complex sequences, severe

accidents and limiting events.

B. LFR CAN MEET THE FOUR GOAL AREAS AND EIGHT SPECIFIC GOALS OF GENERATION IV

The members of the Generation IV International Forum (GIF) Provisional System Steering Committee (PSSC) have evaluated technology options and support the LFR based on its promise in meeting the Generation IV objectives. In particular, the GIF PSSC members have evaluated the two selected small and medium-size LFR conceptual designs by considering the ***four goal areas*** and ***eight specific goals*** of Generation IV.

The main features that the members have identified in order to achieve the GEN IV goals are discussed below and summarized in Table 4. These features are based either on the inherent features of lead as a coolant or on the specific designs to be engineered for both LFR projects.

Sustainability

- Resource utilization. Because lead is a coolant with very low neutron absorption and moderation, it is possible to maintain a fast neutron flux even with a large amount of coolant in the core. This allows an efficient utilization of excess neutrons and reduction of specific uranium consumption. Reactor designs can readily achieve a breeding ratio of about 1, and long core life and a high fuel burnup can be achieved.

- Waste minimization and management. A fast neutron flux significantly reduces waste generation, Pu recycling in a closed cycle being the condition recognized by GEN IV for waste minimization. The capability of the LFR systems to safely burn recycled minor actinides within the fuel will add to the attractiveness of the LFR.

Economics.

- Life cycle cost. The cost advantage features of the LFR must include low capital cost, short construction duration and low fuel and production cost. The economic utilization of MOX fuel in a fast spectrum has been already demonstrated in the case of the SFR, and no significantly different conclusion can be expected for the LFR except for improvement due to the harder spectrum.

Because of the favorable characteristics of molten lead, it will be possible to significantly simplify the LFR systems in comparison with the well known designs of the SFR, and hence to reduce its overnight capital cost, which is a major cost factor for the competitive generation of nuclear electricity.

A simple plant will be the basis for reduced capital and operating cost. A pool-type, low-pressure primary system configuration offers great potential for plant simplification.

The use of in-vessel Steam Generator Units (SGU's) and the consequent elimination of the intermediate circuit, typical of sodium technology, are expected to provide competitive generation of electricity in the LFR. This approach is possible because of the absence of fast chemical reactions between lead and water, although the SG tube rupture accident (i.e., pressure waves inside the SGU) must be considered in the design. The configuration of the reactor internals will be as simple as possible. The very low vapor pressure of molten lead should allow relaxation of the otherwise stringent requirements of gas-tightness of the reactor head and possibly allow the adoption of simple fuel handling systems.

Corrosion by molten lead of candidate structural steels for the primary system will be minimized by limiting the core outlet temperature. Considering that there will be no intermediate circuit to degrade the thermal cycle and that the expected core inlet temperature of about 400°C is relatively high, the adoption of a high-efficiency water-steam supercritical cycle is possible. Additionally, a supercritical carbon dioxide Brayton cycle energy conversion system can be considered.

- Risk to capital. For small, transportable systems, a limitation to the risk to capital results from the small reactor size. In addition, and with particular relevance to the moderate- or large-size central station system, a reduction in the risk to capital results from the potential for removable/replaceable in-vessel components.

Safety and Reliability

- Operation will excel in safety and reliability. Molten lead has the advantage of allowing operation of the primary system at low (atmospheric) pressure. A low dose to the operators can also be predicted, owing to its low vapor pressure and high capability of trapping fission products and high shielding of gamma radiation. In the case of accidental air ingress, in particular during refueling, any produced lead oxide can be reduced to lead by injection of hydrogen and the reactor operation safely resumed.

The moderate ΔT between the core inlet and outlet temperatures reduces the thermal stress during transients, and the relatively low core outlet temperature minimizes the creep effects in steels.

- Low likelihood and degree of core damage. It is possible to design fuel assemblies with fuel pins spaced further apart than in the case of sodium and this allows a large coolant fraction as in the case of the water reactor. This results in a moderate pressure loss through the core of about 1 bar, in spite of the high density of lead, with associated improved heat removal by natural circulation and the possibility of an innovative reactor layout such as installing the primary pumps in the hot collector to improve several aspects affecting safety.

Lead allows a high level of natural circulation of the coolant; this results in less stringent requirements for the timing of operations and simplification of the control and protection systems.

In case of leakage of the reactor vessel, the free level of the coolant can be designed to maintain a level that ensures the coolant circulation through, and the safe heat removal from the core. Any leaked lead would solidify without significant chemical reactions affecting the operation or performance of surrounding equipment or structures.

- ***No need for off site emergency response.*** With lead as a coolant, fuel dispersion dominates over fuel compaction, preventing severe re-criticality. In fact lead, with its higher density than oxide fuel or low-density metal fuel, and its natural convection flow, does not permit fuel aggregation with subsequent formation of a secondary critical mass in the event of postulated fuel failure.

Proliferation Resistance and Physical Protection

- Unattractive route for diversion of weapon-usable material. The use of a MOX fuel containing MA increases proliferation resistance. The use of a coolant chemically compatible with air and water and operating at ambient pressure enhances -***Physical Protection.*** There is reduced need for robust protection against the risk of catastrophic events, initiated by acts of sabotage because there is a little risk of fire propagation and because of the passive safety functions. There are no credible scenarios of significant containment pressurization.

Table 4 - LFR potential performance against the four Goal Areas and the eight Goals for Generation IV.

GEN IV Goal Areas	***Goals for Generation IV Nuclear Energy Systems***	***Goals achievable via***	
		Lead inherent features	***Specific engineered solutions***
Sustainability	Resource utilization.	• Lead is a low moderating medium. • Lead has low absorption cross- section. **Error! Bookmark not defined.** • This enables a core with fast neutron spectrum even with a large coolant fraction.	• Breeding ratio close to 1
	Waste minimization and management.		• Great flexibility in fuel loading including homogeneously diluted MA.
Economics.	Life cycle cost.	• Lead does not react with Water. • Lead does not burn in air. • Lead has a very low vapor Pressure. • Lead is cheap.	• Reactor pool configuration. • No intermediate coolant loops. • Compact Primary System. • Simple design of the reactor internals. • Supercritical steam (high efficiency).
	Risk to capital (Investment protection).		• Small reactor size. • Potential for in-vessel replaceable components
Safety and Reliability.	Operation will excel in safety and reliability.	Lead has: • very high boiling point; • low vapor pressure; • high shielding capability for gamma radiation; • good fuel compatibility and fission product retention.	• Primary system at atmospheric pressure. • Low coolant ΔT between core inlet and outlet.
	Low likelihood and degree of core damage.	Lead has: • good heat transfer characteristics; • high specific heat and thermal expansion coefficient; • core with inherent negative reactivity feedback.	• Large fuel pin pitch. • Decay Heat Removal (DHR) in natural circulation. • Natural circulation cooling (small system). • Primary pumps in the hot collector (moderate- or large- size system). • DHR coolers in the cold collector.
	No need for off site emergency response.	• Lead density is close to that of fuel; no risk of re-criticality in case of core melt. • Lead retains released fission products.	

Proliferation Resistance and Physical Protection.	Unattractive route for diversion of weapon-usable material.	• Lead system neutronics enables long core life.	• Small system features sealed, long-life core. • Use of a MOX fuel containing MA increases Proliferation Resistance.
	Increased physical protection against acts of terrorism.	• Primary coolant chemically compatible with air and water operating at ambient pressure.	• Independent and redundant DHR loops operating in natural circulation.

C. GIF RESEARCH & DEVELOPMENT PLAN FOR THE LEAD-COOLED FAST REACTOR (LFR)

One of the principal purposes of the GIF R&D plan is to identify the priorities of the international GIF community (as represented by the LFR-PSSC) for common or coordinated LFR research, and the proposed R&D approach to achieving them. The intent is to address the needs encompassed in the dual-track approach leading to the development of both a small, transportable system and a moderate- or large-scale central station LFR plant.

The R&D priorities (and planned efforts) focus on the following major topics: system design (already described); fuels development; lead technology and materials; component development; balance of plant; hydrogen production; demonstration. The plan recognizes differences in priorities in each of these categories based on the dual track approach; nevertheless, opportunities for collaboration (and convergence of priorities) exist and should be exploited.

Figure 4 below illustrates the basic concept underlying the LFR R&D plan. It portrays the dual-track viability R&D program leading to a single, combined demonstration facility leading to eventual deployment of both types of systems. The ELSY preliminary design is integral part of the GIF plan as well the associated experimental activity which is part of the GIF Viability R&D

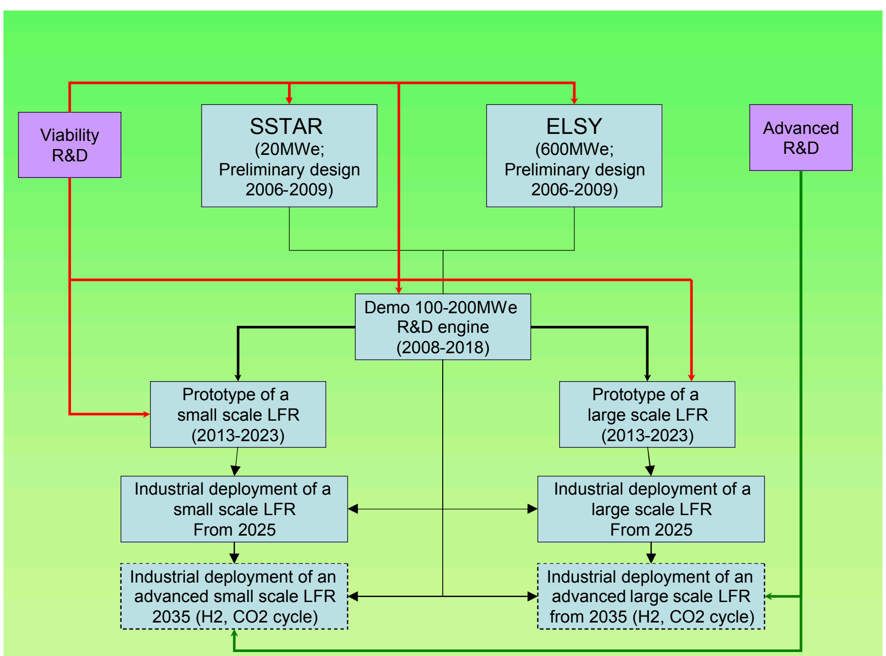

Figure 4- LFR R&D Conceptual Framework and Schedule

C.1 Fuel

Short term

MOX fuel is considered as a reference fuel for short term deployment, because of its availability and state of qualification, not to introduce additional risk of delay for deployment.

Long term

The R&D Plan for the Lead-cooled Fast Reactor (LFR) presents several development lines concerning the fuel to be used in the long run without defining clear priorities for the moment because several inputs are still missing. In particular, the ELSY project will provide at least two type of information:

- the incentive to develop nitride fuels
- the capability of the system to accept fuel containing MA.

General fuel cycle activities, such as fuel dismantling, dissolution, separation, FP confinement re-fabrication are crosscutting with the needs not only for the LFR fuel cycle R&D but also for GFR, SFR, and SCWR (Fast) fuel cycle R&D. Close interactions are anticipated between the LFR System Steering Committee (LFR-SSC) and the Fuel Cycle Project Management Board.

A tentative date for final selection of an advanced fuel can be established around the 2015 as foreseen for SFR.

Because lead has low lethargy, it provides a continuous neutron spectrum from fast to thermal crossing also the resonance energy of LLFP (Long Lived Fission Products) their transmutation can also be envisaged and specific matrix have to be developed, even if this is considered of second priority with respect to fuel development for MA burning.

The candidate LLFPs identified are basically iodine (I-129) and technetium (Tc-99). The fact that iodine I-129 has high solubility, is absorbed weakly in the geological formation and has a long half life of 1.6×10^7 years, suggests that any artificial barrier, will have difficulties in keeping it contained for very long periods.

Consequently, iodine is a major contributor to the risk calculated in performance assessments and can be considered, in a sense, to have high priority as a candidate for transmutation.

C.2 Lead technology and materials

Short term

It is to be noted that Europe has already a large research program in progress or planned on lead that is a substantial part of the overall GEN IV program [7].

Besides the background information acquired with the activities performed in FP5, a large research program will be carried out with the IP EUROTRANS-DEMETRA activity to develop and assess structural materials and Heavy Liquid Metal (HLM, lead and lead-bismuth) technologies. This involves specification and fabrication of reference materials, their characterisation in HLM, and irradiation.

A few R&D activities will be dedicated to the confirmation of key features of ELSY, particularly the applicability of the LBE technology to pure lead.

Furthermore to the creation of additional synergy among the European R&D bodies has been proposed by means of an Integrated Infrastructure Initiative (VELLA), which has the main purpose of:

- Creating a community of Researchers with exchange of staff;
- maximising the value of the existing facilities by stimulating and facilitating access for the associated teams to all of the larger laboratories,
- promoting protocols, standards and a consistent methodology for measurement techniques,
- training of a new generation of nuclear engineers and scientists to become experts in HLM technology.

Materials

As the development of new materials is a very time consuming process, for short term deployment it is necessary to make use, to the maximum extent possible, of available materials by limiting activities to conditions compatible with the known performance limits of these available materials. To establish reactor feasibility, it is necessary to provide a technologically viable structural material capable of withstanding the rather corrosive/erosive operating conditions of a LFR.

Austenitic steels, due to the large database available for such materials, especially those of low-carbon grade, are candidates for components operating at relatively low temperatures and low irradiation fluence, as is the case of the reactor vessel.

Ferritic-martensitic steels appear to be candidate materials for fuel cladding and structures which are under high irradiation flux.

Under the auspices of the ADS activity in Europe, there is an ongoing experimental campaign of corrosion tests in stagnant and flowing LBE under different oxygen activities. From these preliminary results in LBE it can be stated that:

- corrosion rate remains negligible up to 400°C even in reducing environment for the tested ferritic-martensitic steel and stainless steels;
- for exposure times up to 7000 h, it can be concluded that austenitic steels (e.g. AISI 316L) can be employed in LBE with the appropriate control of oxygen activity up to a temperature of 500 °C. Martensitic steels (e.g. T91) can be probably used in this environment up to 550 °C, but for limited time because of the high oxidation rate;
- also as a result of experiments carried out in the past (e.g., at TECLA), mainly with GESA and pack cementation alloyed steels, it was shown that FeAl coating acts as an effective corrosion barrier at temperatures up to 600 °C in LBE with controlled oxygen activity. *It should be noted that task 6.4 of ELSY will complement this need with the exception of validation under fast flux conditions.*

A few comparative corrosion tests show that Pb is less corrosive than LBE for the same conditions of temperature and coolant speed.

From the reliable results of corrosion tests on austenitic and ferritic steels, the development of an LFR with primary system operating temperature less than 500°C and fuel cladding less than 600°C is suggested. The remaining R&D needs consist in the qualification of:

- an austenitic steel and related joint welds operating in pure lead at a temperature of 400-450°C flowing at low speed (below 1 m/s);
- a ferritic-martensitic steel and related joint welds operating in a pure lead environment (flowing at a speed below 2 m/s) and under fast flux irradiation at a temperature of 400-500°C;
- a protective coating for fuel cladding operating in a pure lead environment (flowing at a speed below 2 m/s) and under fast flux irradiation at temperatures ranging from

400°C up to 550-600°C.

- a material for components of a mechanical pump.

Even if a low-temperature primary cycle is selected, a large program of basic technology confirmation is necessary covering several aspects like materials specification and fabricability, materials characterisation in lead, materials characterisation under irradiation, advanced thermal-hydraulics, measurement techniques and system behavior confirmation by means of large-scale integral tests.

Lead Technology development

The main objectives are:

- the physical and chemical characterisation of the lead and compilation and validation of the necessary databases in the parameter range of interest. *It should be noted that tasks 6.1 and 6.2 of ELSY cover this need.*
- the development and validation of a technique for lead purification/conditioning before in vessel filling;
- the development and validation of a technique for in-reactor lead purification with reactor in operation and prevention/control of slag / aerosol formation;
- the development and calibration of instrumentation operating in lead and under irradiation: oxygen sensors, thermocouples, pressure transducer, flow meters, strain gauges, neutron flux, velocity measurement devices;
- the development of techniques for failed fuel detection;
- the development of techniques for in-core instrumentation;
- the development of techniques and instrumentation for in-service inspection (ISI) (mainly for the main vessel surfaces operating in air from the outer wall and the Steam Generator (SG) tubes operating steam side from inside the tubes). For ISI of the main vessel it is necessary to develop ultrasonic sensors operating in a relatively high temperature environment (about 350°C) and a suitable development of small size equipment to reduce, for improved safety, the required gap between main and safety vessel. For ISI of the SG tubes a suitable development is that of small diameter probes to reduce, for improved safety, the diameter of the SG tubes;
- the assessment of lead-fuel interaction phenomena;
- the assessment of activation and fission products (mainly iodine, krypton, xenon) diffusion in lead and release process to the cover gas;
- the assessment of lead aerosol formation above the lead free level as a function of the free level temperature and velocity-field and the type of cover gas.

Reference Materials irradiation studies

The objectives are the characterization of the mechanical behavior of the reference structural materials (including the oxide/coating corrosion barrier under fast neutron spectrum exploring the temperature and dpa ranges as defined in the design.

First tests must be performed to address separate-effects behavior (only fast flux environment) to develop models that will describe the irradiation phenomena and corrosion of the materials. Additional lead irradiation experiments will be aimed at characterising the combined effect on the mechanical and corrosion behavior of the material. ISTC programs will be supported to validate the cladding material under fast flux in lead environment.

Thermal-hydraulics in lead

The objectives are related to basic thermal hydraulic studies for the development of physical models and the validation of numerical tools, useful for the design and the safety analysis. These studies are related mainly to the characterization of heat transfer coefficients, the assessment of the lead/water interactions and the development of measurement techniques.

Large-scale integral tests to characterise the behavior of the main systems are necessary especially for the licensing process. Integral system experiments will reproduce the circulation of a sector of the primary lead coolant in the reactor pool. The normal steady state condition operating transients (e.g., pumping system start-up and core power changes) and incidental transients (e.g., transients from forced to natural circulation) will be evaluated by experiment. The experimental interpretation will be made with thermal–hydraulic calculations and the data will support modelling for transient and safety analysis.

Long term

The high boiling point of lead is a significant advantage for high-temperature operation of the reactor extending the LFR mission towards higher efficiency in energy generation and hydrogen production. Those missions require the development of new materials both for mechanical components and fuel cladding. The development of such materials will be time consuming and will be carried out with a flexible schedule depending on investment resources and technological progress. This long term development program could be started in parallel to the short tem development.

Peculiar is the development of a fuel cladding resistant to high dpa (for increased fuel burn-up) and to high temperature (for increased coolant temperature and power density). Example of a fuel cladding material to be investigated for high-temperature operation could be the ODS with or without surface treatment.

C.3 Component development

The most challenging components for LFR development are the Primary Pumps and the Steam Generators (SG).

Validation of the functional sizing and of materials (mainly the pump impeller) operating in erosion/corrosion environment (lead at high speed) is necessary. Coating of the impeller is probably a suitable technical solution, but no experience exists at present in Europe. Alternative promising materials such as the MAXTHALR (Ti_3SiC_2) should be tested. The natural circulation capability of the primary system eliminates the need of safety-grade pumps which makes easier the development and even makes possible to replace the pumps as soon as new material and technologies are made available. It should be noted that task 6.5 of ELSY covers this need at the preliminary level. Small scale tests and final full flow rate and long term operation tests are necessary.

Even the SG is a component with important innovative features, namely the operation with lead, under a supercritical cycle, and with installation within the Reactor Vessel.

The integration of the SGs inside the vessel is a key feature for economics, and, consequently, the demonstration of the system capability to tolerate a SG tube rupture accident is a requirement for safety. The qualification of a high-conductivity material for SG tubes, such as T91, is beneficial for SG compactness. The SG tube rupture has to be tested at a sufficiently large scale of the SG tube bundle to validate the codes and to optimize the SG geometrical configuration for effective damping of the pressure waves without damaging the reactor core. It should be noted that task 6.3 of ELSY covers this need at preliminary level.

To demonstrate their safety role, an extensive test campaign has to be performed for the validation of control rod mechanisms operating in lead and for the DHR system and associated components.

Qualification is also necessary for fuel handling machines.

C.4 Balance of Plant

Short term

In the short term the main objectives are:

- the definition of the auxiliary systems specific for a lead-cooled reactor and their integration inside the reactor building;
- addressing the specificity of a reactor supported by seismic isolators;
- the definition of a feed water system and of a supercritical steam cycle taking into account the specific requirements of the SG and reactor operational modes; optimization of the thermal cycle for system efficiency.

Long term

In the long term it is expected that significant plant improvements will be necessary for extending LFR missions considering that:

- new materials will be made available for higher temperature operation and increased efficiency;
- supercritical carbon dioxide Brayton cycle energy conversion will be made available and adopted for increased efficiency and reduced BOP cost;
- combined production of hydrogen and electricity will be required.

C.5 Demonstration facility

The results of SSTAR and ELSY evaluations that are expected for the year 2008, particularly with regard to the economic viability of the SSTAR for remote communities and the competitiveness of ELSY for central station power generation beyond the well-known advantages of the fast spectrum, should create the conditions for initiating a significant development program including the necessary step of design and construction of a demonstration facility.

From that date, the research plan (Figure 4) could be focused on the design of a demonstration facility of about 100-200 MWe that could serve both the SSTAR and the ELSY projects, validating lead technology and overall system behavior. The demonstration facility will prove the general strategy to use, to the greatest extyent possible, simple solutions, available MOX fuel, classical materials and to operate the system at low temperature, and particularly important for ELSY, to reduce the technological risks to a minimum. Once the correct operation of an “Easy” demonstration facility is demonstrated, more ambitious options (e.g., advanced fuel and new materials for components) will be addressed, relying also on the ability to replace in-vessel components.

Full power operation of the demonstration facility around the year 2018 could justify the initiation at that date of the construction of industrial prototypes of SSTAR and ELSY, the design of which should be carried out in parallel to the construction of the demonstration facility. The correct operation of the prototype systems will create, in turn, the conditions for international industrial deployment around the year 2025, as foreseen in the GEN IV Roadmap of a reactor for electricity generation complying with all GEN IV requirements.

It is expected that focusing the international efforts, an earlier industrial deployment will be possible in the 2020 time frame whereas more advanced solutions operating at higher temperature for hydrogen production or large-scale plants with CO_2 cycle are much more challenging and will be made available at a later date, probably not before the year 2035.

D. CONCLUSION

The LFR has been identified by the GIF as a technology with great potential to meet the needs for both remote sites and central power stations.

In the GEN IV technology evaluations, the LFR system is top-ranked in sustainability, proliferation resistance and physical protection. It is rated good in safety and economics. Safety is considered to be enhanced by the choice of a relatively inert coolant. The LFR is 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 to be deployable by 2025.

A Lead-cooled Fast Reactor Provisional R&D Steering Committee has been set up in the year 2005 under the auspices of the GIF initiative with members from USA, Japan, South Korea and Euratom.

The approach of the GIF plan is to consider two main technology objectives:

- a small, transportable system of 10–100 MWe, and
- a medium- or large-sized system rated at about 600 MWe.

The European contribution to the development of the LFR is decisive and includes results of Russian ISTC projects, projects under the past FP5 and ongoing FP6 activities on ADS, and the proposed Integrated Infrastructure Initiative (VELLA) and ELSY project. Sixteen European Organizations have, in fact, taken the initiative to present to the European Commission the Specific Targeted Research and Training Project (STREP) devoted to the development of a European Lead-cooled System, the ELSY project; two additional organizations from the US and Korea have joined the project, others, including private investors, are ready to support the initiative. This would create the conditions for advancing the activity beyond FP6.

Considering that significant commonality of R&D can be found between the small, transportable system and the medium-or large-sized system of the two GEN IV approaches, the GIF plan proposes coordinated research with a single demonstration facility that can serve the R&D needs of both approaches while reducing the unnecessary expense of duplicate major test facilities. Full power operation of the Demo around the year 2018 using, ton the greatest extent possible simple solutions, classical materials and operating at relatively low temperature, to reduce the technological risks to a minimum, could also justify the construction, at that date, of industrial prototypes of SSTAR and ELSY. The successful operation of the prototype will create, in turn, the conditions for international industrial deployment around the year 2025 as foreseen in the GEN IV Roadmap of a reactor for electricity generation complying with all GEN IV requirements.

More advanced solutions operating at higher temperature for hydrogen production or large-scale plants with CO_2 cycle are much more challenging and will be made available at a later date, probably not before the year 2035.

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