"LEAD-COOLED FAST REACTOR (LFR) DEVELOPMENT GAPS"

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Abstract

The current paper aims to summarize the ongoing research activities for the development of the ELSY system. Focus is given to the key technical issues and corresponding future R&D activities. Because of several similarities with the Lead Fast Reactor (LFR) the strong synergy with Sodium Fast Reactor (SFR) is also presented.

Lead has a high melting point (327.4°C) and a very high boiling point (1745°C). The high boiling point has a beneficial impact to the safety of the system, whereas the high melting point requires new engineering strategies to prevent freezing of the coolant and blockage of the circulation through the core.

Lead is relatively corrosive towards structural materials especially at high temperatures with a consequent necessity to control its purity carefully. Due to its harsh environment coupled with high energy neutrons effects, an accurate choice of materials is required. Among the components, the fuel cladding material is one of the crucial issues.

An extensive R&D program related to heavy-metal cooled systems was recently initiated in Europe. These efforts, conducted under the of EURATOM projects of the 6th and 7th Framework Programme, are addressing many of the most important issues related to the viability of the LFR.

In the ELSY project, care has been given to the issue of consequences mitigation in case of the Steam Generator Tube Rupture (SGTR) accident. Particularly effort has been spend to reduce the risk of primary boundary pressurization. Priority was also given to the goal of reactor compactness to fulfil economic reasons and to improve the plant performance with respect to seismic loads. The abovementioned goals require large innovation of the primary system configuration and the use of innovative components which need extensive experimental campaigns devoted to qualification and functional confirmation.

The Russian technique based on dissolved oxygen control for formation and self-healing of an oxide layer on the structural steel exposed to LBE has been confirmed as an effective technology to protect the steel from corrosion, although its application to pure lead in large pool configurations requires huge investigations. The operating conditions under which ferritic/martensitic and austenitic steels can be employed have been established. Protective coatings are under development. Their use is actually limited to components subjected to high temperatures or high lead velocity that experience less than 2 dpa. Their use to components (fuel cladding) that experience higher dpa (up to 50-100 dpa) has to be proven by irradiation tests on fuel pin.

Embrittlement and corrosion attack by lead under neutron irradiation are at the beginning of the investigations.

The development of LFR fuel is also discussed. Its connection to the development of SFR fuel is addressed in the last part of the paper.

1. INTRODUCTION

On October 2010 the LFR Provisional System Steering Committee (PSSC) has issued the Lead Fast Reactor (LFR) System Research Plan (SRP) [1]. The committee selected two pool-type reactor concepts as candidates for international cooperation and joint development in the GIF framework: these are the Small Secure Transportable Autonomous Reactor (SSTAR) [2]; and the European Lead-cooled System (ELSY) [3].

In evaluating and planning research for these LFR concepts, the LFR-PSSC has followed the general aims of the Generation IV Roadmap; thus, efforts have focused on design optimization with respect to sustainability, economics, safety and reliability, proliferation resistance and physical protection [4]. Consideration of these factors has guided the identification of the R&D guidelines to bring these concepts to fruition.

The needed research activities are identified and described in the SRP. It is expected that in the future, the required efforts could be organized into four major areas of collaboration. The four areas are: system integration and assessment; lead technology and materials; system and component design; and fuel development.

In Europe, the Strategic Research Agenda (SRA) of the Sustainable Nuclear Energy Technology Platform (SNETP) has selected three Fast Neutron Reactor systems as a key structure in the deployment of sustainable nuclear fission energy. Two of them are cooled by liquid metal: the Sodium-cooled Fast Reactor (SFR) and the Lead-cooled Fast Reactor (LFR). The last one is cooled by helium: the Gas-cooled Fast Reactor (GFR).

In this paper, the key technical issues and the corresponding future R&D activities for LFR are discussed.

2. LEAD TECHNOLOGY

Research and design on the use of the Lead-Bismuth Eutectic (LBE) alloy as a coolant for nuclear reactors was initiated in the early 1950s in Russia for military submarine propulsion. The first LBE- cooled nuclear submarine was put into operation in 1963 and in total 15 reactors have been built including 3 land system reactors, plus one replacement reactor for submarines.

However, LBE, has two major drawbacks. The first one is represented by Bismuth transmutation into highly radioactive polonium by neutron capture, which limits the access to the reactor and requires extensive use of robotic systems. The second one deals with re-crystallisation: LBE undergoes expansion in the solid state which can damage the mechanical structures in case of freezing.

In addition LBE has shown other inconveniences such as formation of solid impurities, “black dust” and macroscopic slag with consequent potential for filter and pipe occlusions, loop malfunctions, and cover gas piping blockage. Recent experiences acquired by ENEA have shown that this does not occur with pure lead [5].

For this reason most of the civil reactor projects developed in the last years are based on pure lead as coolant. Among them, BREST-300 and BREST-1200 have been launched in Russia; ELSY and its evolutions ELFR and the LFR-Demo ALFRED have been proposed in the framework of European projects, and SSTAR in USA. LBE is mainly reserved to experimental reactors mainly because of the lower freezing temperature when compared to lead and for the large power density that can be obtained even at low operating temperature.

Lead (Table 1) is characterized by high melting point (327.4°C) and very high boiling point (1745°C). The high boiling point has a beneficial impact on the safety, whereas the high melting point requires new engineering strategies to prevent its freezing anywhere in the system, particularly during reactor shut down, maintenance and refuelling. Lead, especially at high temperatures, is relatively corrosive towards structural materials Therefore, the accurate selection of the structural materials for the different components and the necessity to control the lead purity represent one the most crucial issues [6].

TABLE 2. MAIN PROPERTIES OF LEAD.

	Atomic weight	Reactivity w/Air and Water	Boiling Point (°C)	Retention of fission products	Melting Point (°C)	Compatibility with structural materials	Opacity	Density (kg/m3) @400°C
Lead	**207,2**	**Low**	**1737**	**High**	**327.4**	**Corrosive**	**Yes**	**10580**

As a result of its **high atomic mass** lead shields γ-rays effectively and elastic scattering of neutrons is characterized by relatively low energy loss while limiting neutron leakage from the inner region of the core and, at the same time, providing an excellent reflection for the neutrons, which escape the core.

The low moderating capability and low neutron capture permits to open up the fuel pin lattice whereby increasing the coolant volume fraction without significant reactivity penalty. Increasing the coolant volume fraction increases the hydraulic diameter for coolant flow through the core with a corresponding reduction of the core pressure loss. As a result of these factors and the thermodynamic properties of lead, natural circulation is effective and can transport a great fraction of the core power.

A core having a large pin spacing and large pin diameter requires a different approach for supporting the fuel pins than that typically employed for SFRs. Thus, a fuel assembly wrapper is not required, and this enables elimination of a seizable amount of steel in the core that would have to be otherwise accounted for. Moreover, without wrapper, the open fuel assembly can be designed akin to that of Pressurized Water Reactors with the advantage of reducing risks of flow blockage as an accident initiator and of cooling the core by convective streams in case of lead freezing inside the SG and/or in the downcomer.

The **lack of fast chemical reaction of lead with water/steam** makes it possible to eliminate the intermediate cooling circuit, which otherwise would be a need, thereby substantially reducing the plant complexity and capital cost while enhancing plant reliability.

The very **high boiling point** of lead allows operation at high temperature which is only limited by the need to preserve the integrity of the steel cladding and structures, but not to avoid risk of coolant's boiling. Lead does not flash, should a leak occur in the primary coolant system boundary: this fact enables the significant advantage of a low-pressure, compact liquid metal coolant system. The present pool-type LFR primary coolant system configuration incorporates all the main primary system components, including the core, the Primary Pumps and Steam Generators (SGs), inside of the Reactor Vessel eliminating the intermediate coolant system.

The lack of risks of coolant flashing or boiling, combined with the reactor/guard vessel configuration, precludes the accidental loss of primary coolant and ensures that adequate core cooling, heat transfer to the steam generators or to in-vessel emergency coolers takes place by natural convection, in case of loss of station service power.

Lead has a **high retention capability of volatile contaminants** and, in particular, appears to form compounds with Iodine and Caesium up to 600 °C. This reduces the source term during postulated accidents, when volatile fission products are released.

The **high density of lead** allows to predict that oxide fuel dispersion at the free coolant's surface would dominate over fuel compaction in case of postulated core melt. This would eliminate any risk of fuel collection at the reactor vessel bottom, making re-criticalities un-probable. A core catcher is therefore not necessary for a LFR.

3. THE ELSY PROJECT EXPLOITS THE PROPERTIES OF LEAD

The ELSY project (Table 2) developed in the 6th Framework Programme of EURATOM has shown for the first time that is possible to design a competitive and safe fast critical reactor using simple engineered features, whilst fully complying with the Generation IV goals of sustainability, economics, safety, proliferation resistance and physical protection.

TABLE 2. MAIN CHARACTERISTICS OF ELSY.

Parameter	ELSY
Power, MWth	1500
Thermal efficiency, (%)	42
Primary coolant	Lead
Primary coolant circulation (at power)	Forced
Core inlet temperature, (°C)	400
Core outlet temperature, (°C)	480
Fuel	MOX, (Nitrides)
Peak cladding temperature, (°C)	550
Fuel pin diameter, (mm)	10.5
Active core dimensions Height/equivalent diameter, (m)	0.9/4.32
Power conversion system working fluid	Water-superheated steam at 18 MPa, 450°C
Primary/secondary heat transfer system	Eight Pb-to-H2O SGs
Fuel column height. (mm)	900
N° Fuel Assemblies (FA)	162
FA geometry	Open square
FA pitch, (mm)	294
N° fuel pins / FA	428
Fuel pins pitch (at 20°C, (mm)	13,9 square
Fuel pins outer diameter, (mm)	10,5
Enrichment, (%wHM)	14.54/17.63/20.61 Pu, three radial zones

The main technical issue of LFR is related to the protection of the integrity of structural materials at high temperature. The thermal cycle that has been therefore purposely selected with 400°C as core inlet temperature - to have sufficient margin above the lead freezing point and to avoid excessive embrittlement of structural material in fast neutron flux - and only 480°C as mean core outlet temperature (Figure 1) to mitigate corrosion, and to take advantages in term of creep and reduced thermal shocks in transient conditions.

The drawback of such a thermal cycle is the need to increase the coolant flow rate which impacts on the primary system dimensions. This is due to the low lead velocities that can be achieved (<2m/s) in order to reduce corrosion and erosion phenomena.

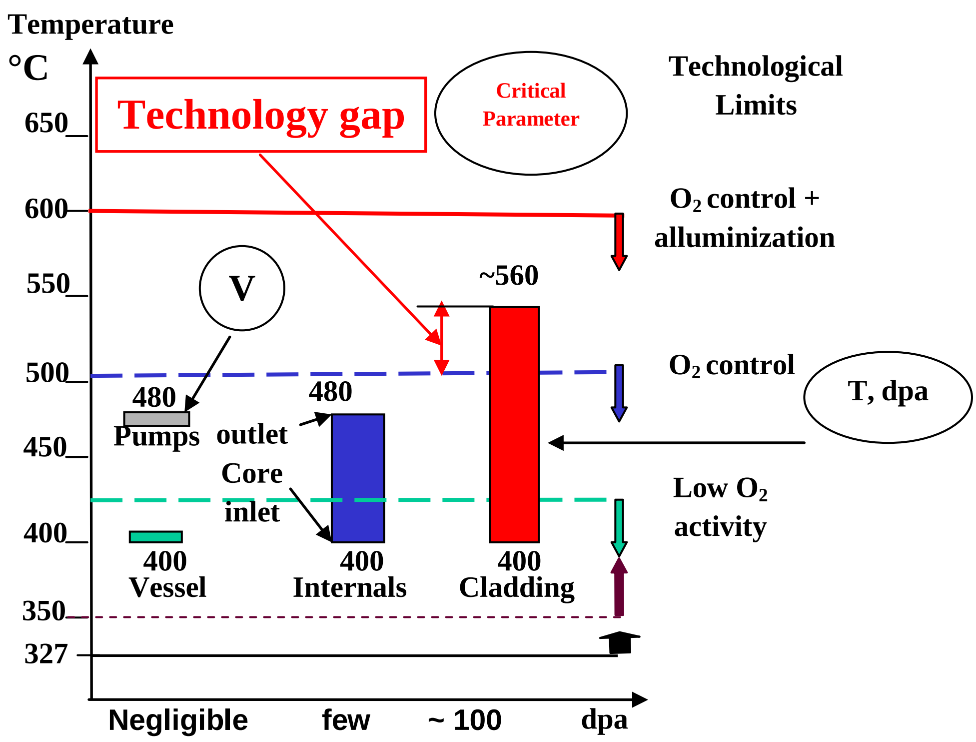


Figure 1- Low operating temperature to limit the technology gaps.

Additionally, the use of a coolant with very high density combined with large primary system, makes the mechanical design challenging with respect to mechanical loads, particularly to seismic loads.

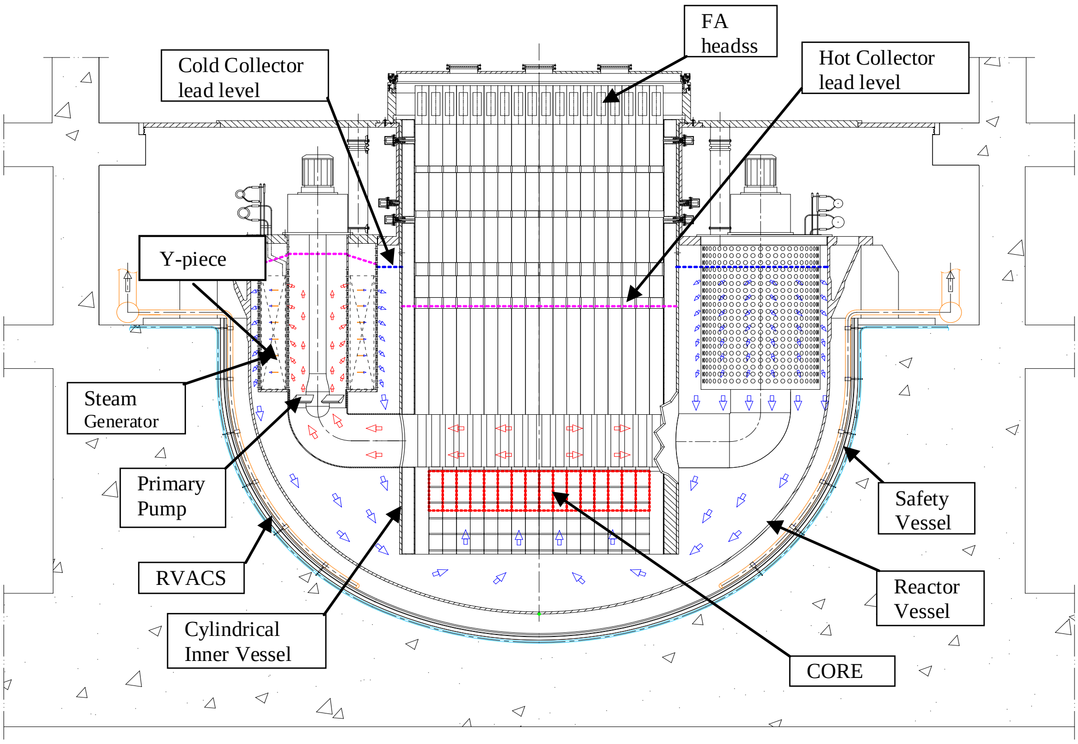
Based on the above mentioned considerations, a large effort has been made to design an innovative primary system as compact as possible, to be accommodated in a short-height reactor vessel, this being a design feature considered basic for a robust design against seismic loads. The result is a reactor with very short vessel (~ 9 m high), whose feasibility is confirmed by the preliminary mechanical analyses. This result, together with the elimination of the intermediate loop, opens the way to the feasibility of a competitive LFR.

Table 3 summarizes the main issues to be addressed and the design provisions proposed to overcome the challenges.

TABLE 3. MAIN ISSUES OF LFR AND SOLUTIONS PROPOSED FOR ELSY.

Issue	**Proposed solution**
Corrosion of structural material in molten lead	Operation at reduced core mean outlet temperature (480°C) No mechanism operating in lead New material for pump impeller Aluminization of heat transfer surfaces
Large mass of lead	Compact primary system Short height reactor vessel
ISI of core support structures in lead	Limitation of support structures immersed in lead
Refuelling at high temperature (400°C) in lead	Handling machines operating in the cover gas plenum
Managing of the SG tube rupture accident	Several preventive and mitigating features
Investment risk of a new technology	All reactor internals can be easily replaced, if needed

The adoption of a pool-type reactor configuration and, most importantly, the installation of a new-design, short-height steam generator (SG) with integrated mechanical pump are the main provisions to achieve the goal of design compactness (Figure 2).



Figure

Figure 2. Primary system configuration of ELSY

The technical risk associated with the corrosive behaviour of lead does not readily permit, with the present corrosion protection technology based on dissolved oxygen, assurance of the ability to achieve the decades-long lifetime of the high-temperature components normally required for nuclear application. The only possible outcome of this issue has been so far the demonstration of the possibility to remove all the primary system components immersed in lead and their replacement with spare components.

Peculiar, innovative solution of ELSY is the cylindrical inner vessel of circular cross section, provided with ducts branching out from its lower part, each duct feeding hot lead to one SG. The cylindrical inner vessel provides also for the lateral restraint of the core, but in contrast to previous solutions, it is not connected to a core support plate, which has been disposed of in ELSY thanks to more advanced solutions. In general, the core support plate constitutes a critical component because it is exposed to fast neutron flux, it is not easy to replace and its in-service inspection and repair results prohibitive. The simple cylindrical inner vessel can be supported in the upper part by the roof and is removable for replacement in case of need.

To be removable, the internals are hung from and supported by the reactor roof. The penetrations through the reactor roof are gas tight-sealed because reactor roof and reactor vessel constitute the primary containment.

The innovative SG conceived for ELSY offers several advantages in term of reactor cost, safety and reactor operability and simplicity of the lead flow path.

The SG tube bundle is composed of a stack of spiral-wound tubes arranged in the bottom-closed, annular space formed by the perforated outer and inner shells of the SG. The inlet and outlet ends of each tube are connected to the feed water header and steam header, respectively, both arranged above the reactor roof. The tube spirals (one spiral for each tube, two spirals per layer), are arranged equally spaced one above the other .

Hot lead flows radially through the perforated inner shell and, once past the tube spirals, through the outer shell. There is no window as a primary coolant inlet port and consequently there is no constraint - typical of the classical design - to locate deep enough the bottom edge of the window to cope with the case of a leaking reactor vessel. In fact, the shell perforations extend below the accidental coolant free level and ensure adequate flow rate for core cooling. Consequently, the SG unit can be positioned at a higher level in the downcomer and the reactor vessel shortened accordingly.

The installation of SGs inside the reactor vessel is a major challenge of a LFR design which includes the need of a sensitive and reliable leak detection system and of a reliable depressurization and isolation system. In ELSY, careful attention has been given to mitigation of the consequences related to SG tube rupture in order to reduce the risk of pressurization of the primary boundary. In this way, innovative provisions have been conceived which make the primary system more tolerant of the SG tube rupture event.

Scoping calculations of the main thermal-fluid-dynamics as well as structural aspects of the SG tube rupture accident were performed adopting appropriate codes (SIMMER, MARC and DYTRAN). The cover gas plenum above the lead plays an important role as an expansion volume to dump both the pressure waves and the pressure surge inside the SG.

The installation of the pump impeller at an immersion depth of a few meters ensures a realistic net positive suction head and, consequently, a short pump shaft is enough to connect the pump impeller to the pump motor located on the reactor roof. No supporting bearing in lead is necessary.

The LFR, in a similar manner to the LWR, relies on the water-steam system to remove decay heat. A reliable DHR system is the reactor vessel air cooling system, the pipes of which are located in the reactor pit between the reactor vessel and the anchored safety vessel.

Unfortunately, reactor vessel air cooling systems by themselves have limited power and can be used only in small-size reactors to enable DHR, the reactor vessel outer surface of which is relatively large. For a large power reactor it is necessary to install additional loops equipped with dip coolers, a safety related DHR system called hereinafter the Direct Reactor Cooling System. This system is comprised of loops which operate with storage water.

Stringent safety and reliability requirements of the safety-grade DHR system are achieved by redundancy and diversification; the system is provided with steam condensers on steam loops branched off the steam-water system piping.

The reference core [8] consists of an array of open Fuel Assemblies (FAs) each containing fuel pins in square pitch surrounded by reflector-assemblies, a configuration that presents reduced risk of coolant flow blockage.

Preliminary analyses confirm a pseudo-exponential build up or decrease of MA toward the equilibrium content of about 310 kg, which means roughly 0.9% HM or 4.8% Pu. This can be considered compatible with safety requirement. ELSY is conceived as “adiabatic” reactor, meaning that it has a Conversion Ratio of about 1 and it burns self-generated Minor Actinides.

The FAs, whose weight is supported by buoyancy and hence does not require a support grid at their bottom head, are fixed at their upper head in the cold gas space, well above the lead surface. This avoids the classical issue of a tricky in-service inspection of a core support grid immersed in lead environment. The lower part of each FA consists of the fuel pin bundle with structural grids similar to the grids of a PWR. The FA foot is free from mechanical supports (no core grid of classical design) except for the radial touching with adjacent FAs and dummy assemblies which are brought into contact to create a packed bundle.

The upper part of the FA is peculiar to this novel ELSY design, because it extends well above the fixed reactor roof, and the FA heads are directly accessible for handling from the above reactor enclosure.

The FA are withdrawn from, and driven into the core using a simple handling machine that operates in the cover gas at ambient temperature, under sight control. This is a typical appropriate fuel handling strategy to overcome the lead opacity and taking advantage of its high melting point.

Table 4 summarizes the key issues, proposed strategies and R&D needs based on the ELSY design.

TABLE 4. SUMMARY OF KEY ISSUES, PROPOSED STRATEGIES AND R&D NEEDS

General issue	Specific issue	Proposed strategy/needed R&D
Lead technology	Lead purification	Technology for the purification of large quantities of lead to be confirmed.
	Oxygen control.	Extend oxygen control technology to pure lead for pool reactors.
Materials resistant to corrosion in lead.	Material corrosion	Selection of a low core outlet temperature for initial reactor design.
		Development of new materials for service at temperatures up to 650°C
	Reactor vessel corrosion.	Vessel temperature limited by design to about 400°C.
	Fuel cladding	15-15 Ti Selection of aluminized surface treated steels for cladding
	Reactor internals	Materials protected by oxygen control
	Heat removal	Confirmation of the suitability of aluminized steels for steam generator to avoid lead pollution and heat transfer degradation.
	Pump impeller	Test of innovative materials at high lead speed
Potentially high mechanical loading	Earthquake	Reactor building built with 2D seismic isolators+short vessel design.
	SGTR accident	Prevention by design of: - steam entrainment into the core; - reactor vessel pressurization; - pressure wave propagation across the primary system.
Main safety functions	Diversified, reliable, redundant DHR	Use of both atmospheric air and pool water.
	Diversified, reliable, redundant reactor shut down system	Confirmation of operation of diversified solutions is needed.
Special operations	Refuelling	Innovative solutions are proposed for ELSY with refuelling machine operating in gas.
	ISI & Repair	Reduction by design of the need for ISI. Operation of devices at ~400°C in lead needs to be verified.
Fuel and core design	Fuel selection	Use of MOX for LFR short-term deployment. MA bearing fuel and high burn-up fuels to be developed in synergy with SFR.
	Lead-fuel interaction	To be assessed
	Failed fuel detection	New solutions to be investigated.
	Needs of appropriate computer codes.	Verification and validation of CFD codes, thermal hydraulic SC, and neutronic codes for LFR application. Development verification and validation of correlations and models that deals with lead chemical behaviour

4. STATUS OF LFR CORE DESIGN IN ELSY AND ALFRED

LFR conventional MOX fuel is designed on the basis of selected criteria and boundary conditions that include thermal hydraulic, neutron kinetic and chemistry. Table 5 reports the main parameters applied in ELSY. ALFRED differentiates in the thermal power (130 MWth).

TABLE 5. LFR BOUNDARY CONDITIONS AND FUEL DESIGN CRITERIA.

Boundary Conditions
Lead core inlet/outlet temperature: *400 / 480 °C* Lead velocity maximum allowable: *<2m/s* Power: *600 MWe (efficiency 40%)* High burn-up: *100 MWd/kgHM* Neutron flux: *2.4 *10^15 [n/cm^2/s]*
Design Criteria
Max allowed peak linear power *32kW/m* Max clad and fuel temperatures in NO *560 °C and 2100 °C* Max clad temperature in accident conditions *1007°C* Max pressure in gas plenum *50 bar* Peak clad damage *max {100 MWd/kg-HM, 100 dpa}* Max clad lead-side corrosion layer *100µm*

The ELSY core is made up of 162 open square Fuel Assemblies (FAs) arranged in three radial zones: 56 FAs in the inner zone with a Pu enrichment of 14.0%, 50 FAs in the intermediate zone 17.0% enriched and 56 FAs in the outer zone 19.9% enriched. The material selected both for the cladding and the FA structures was T91 F/M steel. The fuel cycle management tentatively adopted is 5 years fuel residence time and the refueling of ¼ of the core each 1.25 years. The FA consists of 428 fuel pins arranged in a 21x21 square lattice. The results of the fuel rod conceptual design are summarized in TABLE **6**6.

TABLE 6-ELSY FUEL ROD CONCEPTUAL DESIGN.

Parameter	**Value**
Fuel type	MOX
Fuel column height	900.00 mm
Fuel pellet diameter	9.00 mm
Fuel hollow diameter	2.00 mm
Pellet density	10.48 g/cm^3
Pellet height	12.00 mm
Pellet Oxygen/Metal ratio	1.97
Clad inner diameter	9.30 mm
Gap Clad/Pellet	0.15 mm
Clad outer diameter	10.50 mm
Clad thickness	0.60 mm
Fuel rod pitch	13.90 mm

Hexagonal, wrapped with spacer grid FA has been selected for ALFRED core. It relies on proven MOX fuel. At present, only the maximum enrichment (30%), the active height (600 mm), and the inner diameter of the annular pellets (2 mm) have been definitively fixed. The core design is still ongoing. Preliminary calculations pointed out that 171 FAs with 2 enriched radial zones pursuit a good flux radial flattering.

5. FUEL CLADDING

The main function of the cladding is to contain the fuel. In the defense-in-depth concept, it represents a barrier against the release of the fission products. The qualification of the cladding material is one of the most crucial issues in LFR technology [13]. Historically, the main limiting factor is related to cladding swelling. The swelling acceptability limit may range from 3% to 6%. However, the cladding qualification requires the verification of several multidiscipline technical criteria at the burn-up of interest [9] The following criteria have been developed at LA-NL [10], they are basically the same of Ref. 8. They are numbered in order of importance based on a SFR that would experience 200 dpa (the final burn-up affects the ranking).

- Radiation resistance: The material must be tolerant up to a given dpa under a given range of operating temperatures of swelling, irradiation creep, and embrittlement effects.
- Mechanical properties: tensile properties (strength, ductility), fracture toughness and fatigue resistance should enable to contain the fuel during irradiation and discharge.
- Fuel-clad compatibility: the cladding must be resistant to chemical interaction with the fuel during irradiation to prevent clad failure.
- Fabrication and joining: the material must be able to be fabricated into component parts and joined to contain the fuel during the irradiation and during discharge. Welded joints mechanical properties should be known under the effect of irradiation.
- Clad-coolant compatibility: the material must be tolerant of the coolant, with minimal corrosion rate.
- Reprocessing: the cladding must be dissolvable to enable fuel reprocessing.
- Thermal properties: thermal conductivity and thermal expansion are important to maintain desired fuel temperatures during irradiation. No significant variation must occur as result of irradiation.
- Neutronic properties: low neutron absorption is required.
- Material availability: enough raw material must be readily available.

Before claddings being used to design a reactor core, the material should overpass nuclear qualification. The cladding qualification process involves several steps that start with the irradiation of specimens and should end with the irradiation of fueled rods and single FAs.

In the past decades, extensive R&D work has been devoted to austenitic stainless steels by all the countries developing FR [11]. This material, has by far the largest cumulated experience. At the beginning, AISI-316 (Europe and Japan) and AISI-304 (US) were selected as cladding materials. The experimental irradiations evidenced that these materials experience un-acceptable swelling over 50 dpa limiting the fuel rod residence time at low burn-ups.

The national programmes in France and Germany led to the selection of titanium-stabilized (0,4-0,5%) cold worked (15-20%) austenitic steels as reference candidates, namely the French 15-15 Ti alloy and the German DIN 1 4970 alloy. For these cladding materials (or equivalent materials) extensive irradiation results are available in Europe, US Russia and Japan [11]. The functional limit is basically related to irradiation swelling.

In the framework of GEN-IV, the SFR prototype ASTRID is expected to rely on 15-15 Ti (about 130 dpa). The R&D will be devoted to the long term qualification of advanced austenitic Ti+Nb stabilized and ODS advanced steels (200dpa of final goal) [12].

Ferritic/martensitic alloys are being investigated and developed as cladding materials or ducts, particularly in France, the former USSR and the USA. This kind of materials generally highlight excellent swelling resistance coupled with poor creep resistance that had generally limited its use as wrapper material. The highest exposure doses were reached with FFTF oxide fuels at limited peak cladding temperature (600°C), with a record level of about 200 dpa, without cladding failure.

An example of assessment of the qualification status was carried out at LA-NL comparing HT9 and T91 on the basis of the reported criteria. The investigations revealed the substantial lack of irradiation data on fueled rods to start the licensing of a core that relies on T91 as cladding material. Welded joints irradiation data are also poor as well as irradiation creep databases at high dpa.

It is because of these reasons that MYRRHA and then ALFRED up-dated configurations rely on 15-15 Ti cladding. On the other hands, this introduces the necessity to extend the R&D efforts into the field of lead and LBE compatibility with proven cladding materials.

6. CONCLUSIONS

The LFR systems under consideration offer great promise in terms of the potential for providing cost effective, simple and robust fast reactor concepts that are essential to long-term sustainability of the nuclear energy option.

Recent efforts, particularly in the development of the ELSY concept, have gone a long way toward verifying the advantages of lead cooled systems. Clearly, additional work needs to be done, but overall, the prospects continue to appear very positive.

ELSY aims at demonstrating the possibility to design a fast reactor using simple engineered technical features, whilst fully complying with the Generation IV goals of sustainability, economics, safety, proliferation resistance and physical protection.

The elimination of the Intermediate Cooling System and the compact and simple primary circuit with all internal components removable, are among the features to assure reduced capital cost and construction time, competitive electric energy generation and long-term investment protection.

Molten lead has also the advantage of allowing operation of the primary system at atmospheric pressure.

Despite the high density of lead, the pressure loss can be kept pretty low (about one bar across the core for a total of about 1.5 bar across the whole primary system) because low neutron energy losses in lead allow for a larger fuel rods pitch. This provides for significant natural circulation of the primary coolant, which results in a suitable grace time for operation and simplification of control and protection systems.

The use of a coolant chemically inert with air and water and operating at atmospheric pressure greatly enhances Physical Protection.

Corrosion of structural materials in lead is one of the main issues for the design of LFRs. A larger effort has been dedicated to short/medium term corrosion experiments in both stagnant and flowing LBE. A few number of experiments have been carried out in pure Pb resulting in a lack of knowledge particularly on medium/long term corrosion behaviour in flowing lead. Experiments confirm that corrosion of steels strongly depends on the operating temperature and amount of dissolved oxygen. It has been demonstrated that generally, in the low temperature field, e.g. below 450°C, and with an adequate dissolved oxygen activity, ferritic/martensitic and austenitic steels build up a stable oxide layer which behaves as a barrier against leaching of steel elements providing thereby effective corrosion protection.

Corrosion resistance of structural materials can be enhanced by FeAl alloy coating. At the present status of development of the corrosion protection technology, near-term deployment of the LFR is possible only by limiting the mean core outlet temperature to around 500°C. The extension to higher temperatures offered by the high boiling point of lead will be exploited only in the longer term after successful qualification of new, high-temperature materials such as ODS steels, ceramics and refractory metals and relevant corrosion protection techniques.

The qualification of the cladding material is one crucial issue in LFR technology and for short-term deployment it will rely as much as possible on the Sodium Fast Reactor (SFR) fuel development. For this reason, in spite of some scoping studies on the use of F/M steels, the present LFR European projects are based on the use of 15-15 Ti austenitic steels. High burn-up reactors are not expected to be licensed in the short-medium term relying on new materials as cladding. This is due to the complexity and duration of the qualification process itself and to present un-availability of experimental material testing reactors.

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List of abbreviations

ALFRED	Advanced Lead Fast Reactor European Demonstrator
ASTRID	Advanced Sodium Test Reactor for Industrial Demonstration
CR	Conversion Ratio
DHR	Decay Heat Removal
ELFR	European Lead-cooled Fast Reactor
ELSY	European Lead cooled System
FA	Fuel Assembly
FFTF	Fast Flux Test Facility
F/M	Ferritic / Martensitic
GFR	Gas cooled Fast Reactor
HM	Heavy Metal
LBE	Lead-Bismuth Eutectic
LEADER	Lead-cooled European Advanced DEmonstration Reactor
LFR	Lead cooled Fast Reactor
LWR	Light Water Reactor
	MYRRHA Multi-purpose hYbrid Research Reactor for High-technology Applications
MOX	Mixed OXide fuel
ODS	Oxide Dispersion Strengthened alloy
PSSC	Provisional System Steering Committee
SFR	Sodium cooled Fast Reactor
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SNETP	Sustainable Nuclear Energy Technology Platform
SRA	Strategic Research Agenda
SRP	System Research Plan
SSTAR	Small Secure Transportable Autonomous Reactor