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Lead-Cooled Fast Reactor (LFR) Design: Safety, Neutronics, Thermal Hydraulics, Structural Mechanics, Fuel, Core, and Plant Design

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A Compendium of Reactor Technology

5 Lead-cooled Fast Reactor (LFR) Design: Safety, Neutronics, Thermal-Hydraulics, Structural Mechanics, Fuel, Core and Plant Design

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Abstract

1. Lead-cooled Fast Reactor (LFR) Development

The idea of developing fast spectrum reactors with molten lead (or lead alloy) as a coolant is not a new one. Although initially considered in the West in the 1950s, such technology was not pursued to completion because of anticipated difficulties associated with the corrosive nature of these coolant materials. However, in the Soviet Union, such technology was actively pursued during the same time frame (1950s through the 1980s) for the specialized role of submarine propulsion. More recently, there has been a renewal of interest in the West for such technology, both for critical systems as well as for Accelerator Driven Subcritical (ADS) systems. Meanwhile, interest in the former Soviet Union, primarily Russia, has remained strong and has expanded well beyond the original limited mission of submarine propulsion. This section reviews the past and current status of LFR development.

1.a Lead-Bismuth Eutectic (LBE) for submarine propulsion

Heavy liquid metals (HLM) such as lead (Pb) or lead-bismuth eutectic (LBE) were proposed and investigated as coolants for fast reactors as early as the 1950s (e.g., in the USA). In most cases, with breeding a primary driver, sodium became the preferred choice in the sixties, due to the higher power density achievable with this coolant, which resulted in lower doubling times, an important objective at that time. However, major contributions in the development of lead technology were carried out by Soviet (and then Russian) scientists and industries who have actively pursued lead-cooled reactor technology for more than 50 years.

In the early 1950s in the Soviet Union, research and design of the use of lead-bismuth alloy as the coolant for nuclear reactors was initiated by Academician A. I. Leipunsky at the Institute of Physics and Power Engineering (IPPE) in Obninsk.

The principal objective of these efforts was the design and construction of nuclear reactors for submarine propulsion.

The first of these systems, a 70 MWth 27/VT land prototype reactor, achieved criticality and started full power operation at IPPE in 1959. In 1963, the first nuclear submarine with a heavy liquid metal cooled reactor was put into operation. It was designated "Project 645, Submarine K-27, NATO designation November class K-27 variant" and utilized two 73 MWth reactors. Beginning in 1971, two new series of nuclear powered submarines termed "Projects 705 and 705K, NATO designation Alfa class" were put into operation. Both of these series utilized a single 155 MWth reactor. The distinction between the two was based on their steam supply systems, one type of which was designed by the Experimental Design Bureau of Machine Building (OKBM) and the second was designed by the Experimental Design Bureau "Gidropress" (OKB Gidropress). In total, seven nuclear submarines of the Project 705/705K type were constructed following the original single submarine of the "Project 645" type. In addition, a second land-based prototype designated the KM-1 and mainly supporting Project 705K was put into operation at the A. P. Aleksandrov Scientific Technical Research Institute (NITI) in Sosnovy Bor in 1978.

An extensive research and development program focusing on HLM coolant technology and materials, was carried out with emphasis on the chemical control of the liquid metal to avoid the possibility of plugging due to the formation of slag and to enhance corrosion resistance of internal components made from steels specifically developed for such service.

1.b The Russian design for civilian fast reactors cooled by heavy liquid metals

In the 1990s, there was a renewal of interest in Russia concerning lead and LBE as coolants for civilian fast reactors. The lead-cooled BREST (the Russian acronym for Pb-cooled fast reactor) [1] concept developed beginning in the early 1990s is the most widely known; in addition, the Russians have placed considerable effort in the development of the LBE-cooled SVBR (the Russian acronym for lead-bismuth fast reactor) concept.

1.b (i) The BREST 300.

BREST-300 is designed as a multi-purpose reactor; it produces electric power, consumes and produces plutonium, produces radioisotopes for industry and medical applications, and transmutes long-lived fission products and actinides generated in reactor operations.

The main operating mode of this reactor system is base-load power production, although operation at reduced power levels is also anticipated.

It has a semi-integrated, multi-compartment metallic vessel (with characteristics of both pool-type and loop-type cooling systems). The vessel, 19 m in height, has a diameter of 5.5 m at the bottom and 11.5 m at the top. The wide upper part of the reactor vessel is separated from its narrow central part by a barrel that forms an annular chamber, outside the central part of the vessel. In this semi-integral arrangement, the Steam Generator (SG) and the main circulating pumps are placed in the annular chamber, outside the central part of the vessel.

The core is loaded with wrapper-less fuel assemblies of square cross section. The fuel assembly lattice has 121 square cells of which 114 are taken up by fuel rods and seven by guide tubes. The height of the fuel pellet column in the core is 1.1 m and the cover gas plenum s 0.9 m high. The content of plutonium and minor actinides (MA) is 13 wt.%.

The fuel cladding consists of a thin-walled tube of 12% chromium-ferritic-martensitic steel. It has high corrosion resistance to lead, limited swelling and satisfactory temperature dependence of strength and creep.

The outer diameters of the tubes to be used as cladding in the central, middle and peripheral regions of the core are 9.1 mm, 9.6 mm and 10.4 mm respectively.

1.b (ii) The SVBR-75

The SVBR-75 was designed as a modular compact unit to be installed in the Steam Generator (SG) compartments of shut down VVER-440 type reactors.

The main characteristics are [2]:

- Pool-type reactor.
- Two-loop system for Decay Heat Removal (DHR) using natural circulation.
- Guard vessel.
- Fuel subassemblies without wrapper.
- SG with saturated steam generation.
- Low-speed gas-tight motor of less than 500 kW power for main circulating pumps.
- The ability to repair and/or replace all internal components of the reactor.
- Subassembly by subassembly refuelling of the whole core at a time.
- Multi-fuel capability (UO2, MOX with MA, nitrides fuels) with the same reactor design.

The main plant parameters are:

• Thermal power (nominal), MW.	280
• Steam capacity, t/h.	580
• Steam pressure (saturated), MPa.	9.5
• Feedwater temperature, °C.	240.9
• Primary coolant flow rate, kg/s.	11760
• Primary coolant temperature, inlet/outlet, °C.	320/482

• Core dimensions: diameter/height, m.	1.645/0.9
• Number of fuel pins.	12114
• Number of control rods.	37
• Mean power density in the core, MW/m^3.	140
• Mean linear load of the fuel element, kW/m.	24.3
• Refuelling interval, year.	8
• Core charge, (UO2) with uranium: mass, kg/enrichment, %.	9144/16.1
• Number of main circulating pumps.	2
• Lead-bismuth coolant volume in the reactor, m^3.	18
• Reactor outline dimensions: diameter/height, m.	4.55/7.55

1.c Heavy liquid metal cooled accelerator driven subcritical (ADS) systems

The features and the associated technologies of heavy liquid metal coolants inspired several projects in the emerging field of ADS, and in particular lead and LBE have been considered as both coolants and neutron spallation targets for several such energy amplification projects under development in the USA, Europe, Japan and the Republic of Korea since the mid-1990s.

At the Korea Atomic Energy Research Institute (KAERI) and Seoul National University (SNU) in the Republic of Korea, ADS systems have been developed since 1997 in order to explore proliferation-resistant and safe transmutation technology. One such accelerator driven system (ADS) named HYPER (HYbrid Power Extraction Reactor), is intended primarily for transmutation of long-lived nuclear wastes. HYPER uses LBE as both coolant and spallation target.

In Japan, at the Japan Atomic Energy Research Institute (JAERI), an ADS with the thermal power of 800 MW has been designed, where 250 kg of Minor Actinides and some Long-Lived Fission Products (LLFP) can be transmuted annually. R&D has been conducted on ADS using LBE as a spallation target and coolant.

At SCK_CEN, Belgium, since 1997, studies in the field of LBE technology have been carried out for the Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA) project, aimed at the development of a research subcritical reactor driven by an accelerator, where LBE is used as spallation target and coolant.

The MYRRHA design has merged since 2005 with the European project IP-EUROTRANS which includes also the detailed design of the associated linear proton accelerator and a generic conceptual design of the European Facility for Industrial Transmutation (EFIT) in which pure lead is used as the core coolant and spallation material. EFIT is loaded with U-free transmutation-dedicated fuel.

1.d The LFR in Generation IV.

The Generation IV (GEN IV) Technology Roadmap [3], prepared by Generation IV International Forum (GIF) member countries, in 2002 identified the six most promising advanced reactor systems and related fuel cycles, and the R&D necessary to develop these concepts for potential deployment. Among the promising reactor technologies being considered by the GIF, the LFR has been recognised 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 it uses a closed fuel cycle, and top-ranked in proliferation resistance and physical protection because it employs a long-life core. It was rated good in safety and economics. 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 forecast 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 reprocessed LWR spent fuel and as a burner/breeder with thorium matrices.

In 2007, the GIF LFR Provisional System Steering Committee (PSSC), after evaluation of current international initiatives in the field, prepared a draft System Research Plan (SRP) for the Lead-Cooled Fast Reactor with molten lead as the reference coolant and lead bismuth as a backup option. Figure 5.1 below illustrates the basic approach being recommended in the LFR SRP. It portrays the dual track viability research program with convergence to a single, combined demonstration facility (demo, also called Technology Pilot Plant - TPP) leading to eventual deployment of both types of systems. The dual track approach is based on the development of the Small Secure Transportable Autonomous Reactor (SSTAR) and the European Lead-cooled System (ELSY) reactor projects which represent two potential applications of the LFR.

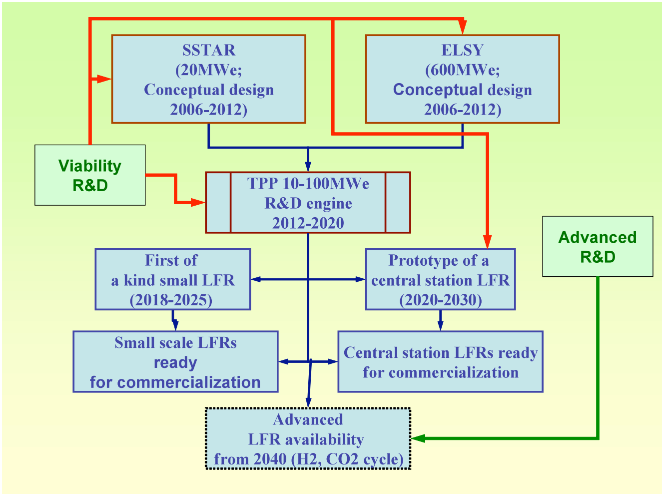

Fig. 5.1 - LFR GIF-SRP Conceptual Framework.

SSTAR, whose development is performed under the U.S. Department of Energy Generation IV Nuclear Energy Systems Initiative, is a small natural circulation fast reactor of 20 MWe/45 MWth, that can be scaled up to 180 MWe/400 MWth. Key technical aspects of SSTAR are the use of lead (Pb) as the coolant and a long-life sealed core in a small, modular system. The compact active core operates for a very long time (15-30 years) without refueling, and the fuel is either retained in the reactor vessel for recycle or removed as a single cassette during refueling and replaced by a fresh core.

ELSY, whose development started in 2006 with the support of the Framework Program 6 (FP6) of Euratom, aims at demonstrating the design of a competitive and safe fast critical reactor using simple engineered technical features. The use of compact in-vessel steam generators and of a simple primary circuit, with possibly all internals being removable, are among the reactor features for competitive electric energy generation and long-term investment protection.

Besides Russia, USA and Europe, LFR studies are also being performed in Japan and Korea.

In Japan, the relevant activities are associated with each of the key research organizations including the Japan Atomic Energy Agency (JAEA), the Central Research Institute of the Electric Power Industry (CRIEPI) and the Tokyo Institute of Technology.

At JAEA a LBE-cooled fast reactor design and the related fundamental corrosion experiments were carried out within the framework of the “Feasibility Study on Commercialized Fast Reactor Cycle Systems” from 1999 to 2005. Experimental studies to solve the corrosion problem have been carried out since 2001. At the early stage of this study, the maximum cladding temperature was set to 650 °C, and then changed to 570 °C based on the results of experimental studies. As a result, the core inlet and outlet coolant temperatures are 285 °C and 445 °C respectively. The specific gravity of LBE is twelve times that of sodium. This property affects structural integrity, a particular concern for the very high seismicity in Japan. According to the feasibility study, it is estimated that the LFR plant size in Japan would be limited to less than medium-scale size of around 750 MWe, even with adoption of 3D seismic isolation.

At CRIEPI the LBE-cooled fast reactor is considered one of the candidates for the next generation of nuclear reactors. CRIEPI started its LBE studies following the proposal of an innovative steam generator for the sodium cooled fast breeder reactor with direct contact heat transfer between LBE in the intermediate loop and water.

To evaluate the heat transfer performance of LBE in the intermediate loop and the two phase flow characteristics of LBE, water and steam, the "CRIEPI Pb-Bi Test Loop on Thermal hydraulics" was constructed in 1997. The results of the heat transfer performance around steam generator tubes, and the performance of gas lift pumps for LBE were tested in the loop and presented in [7].

To clarify the corrosion characteristics of LBE, the "CRIEPI Static Corrosion Test Facility" was constructed in 2001. The objective of this facility was to understand the corrosion behavior of stagnant LBE at 650 °C on high chromium martensitic stainless steel, a promising candidate structural material for LFRs. A series of corrosion tests were performed jointly by CRIEPI and JAEA [8].

To explore the advantages and disadvantages of lead as a coolant, an LBE-cooled 4S (the Super Safe, Small and Simple reactor, normally a sodium-cooled system), was designed and studied by CRIEPI and TOSHIBA.

The Tokyo Institute of Technology proposed a small long-life fast reactor cooled by LBE, and presented a preliminary design in 1991. Since then, the importance of the Tokyo Tech's study has become gradually widely recognized and as a result, programs were supported to promote LFRs in the following areas:

- Po behavior, treatment, cross-section measurements (FY 1998-2000)
- Corrosion (materials test, oxygen control) (FY 1999-2001)
- CANDLE burn-up (FY2001-2003)
- Steam lift-pump reactor designs and basic research (FY2002-2004)

The Pb-Bi Cooled Direct Contact Boiling Water Fast Reactor (PBWFR), the Steam Lift Pump Type LFR (SLPLFR), and the Constant Axial Neutron During the Life of Energy (CANDLE) reactor are the main recent activities in the area of reactor design.

In the PBWFR, direct contact boiling provides significantly higher heat transfer. The PBWFR electric power is 150 MW. The design limit of the cladding temperature is 650 °C. The LBE core outlet temperature is 460 °C. The LBE temperature rise across the core is 150 °C. The conditions of the secondary coolant steam are the same as those of conventional BWRs. The PBWFR plant is equipped with a Reactor Vessel Air Cooling System (RVACS) a Primary Reactor Auxiliary Cooling System (PRACS), and an auxiliary water supply tank to cope with loss of feedwater. Hydrogen is dissolved in the feedwater at a concentration of 100 – 500 ppb to keep the oxygen concentration in the LBE coolant around 10^{-5} wt.%.

The SLPLFR reactor concept has SGs in the reactor vessel, and sub-cooled water is injected into LBE above the core at a low flow rate. The resulting steam condenses in a dedicated heat exchanger, which serves also as the re-heater of the feedwater. In comparison with the PBWFR, the SLPLFR is expected to have higher thermal efficiency with higher LBE temperature, lower pressure in the primary loop, and no Po or LBE droplet contamination in the turbines.

For the CANDLE reactor, the neutron flux shape and the nuclide and power density distributions remain constant but progresses in an axial direction during the core lifetime. The solid fuel is fixed at each position and no movable reactivity control mechanisms are required. The change of excess reactivity during burn-up is theoretically zero for ideal equilibrium conditions. The core characteristics, such as power feedback coefficients and power peaking factors, do not change over the operational life. Since the k-infinity of replacement fuel is less than unity, the transport and storage of such fuels is easy and safe. Application of this burn-up strategy to LFRs with metallic or nitride fuels enables the following excellent characteristics: initial fissile material is required only for the nuclear ignition region of the initial core, and only natural or depleted uranium is required for the remaining region of the initial core and for succeeding cores. The average burn-up of the spent fuel is about 40%; that is equivalent to 40% utilization of the natural uranium without reprocessing or enrichment.

The Korean LFR Program has two main objectives:

- a technology development requirement for nuclear waste transmutation;
- a new electricity generation unit development requirement to match the needs of developing nations and especially remote communities without major electrical grid connections.

To meet the first goal, the PEACER (Proliferation-resistant Environment-friendly Accident-tolerant Continuable-energy Economical Reactor) development was initiated with the objective of developing a system to transmute long-lived fission products in the spent nuclear fuel into short-lived low-intermediate level waste.

For the second goal Korea initiated the development of the BORIS (Battery Optimized Reactor Integral System) reactor system that is an integral-type optimized fast reactor with an ultra long-life core coupled with a supercritical CO_2 Brayton cycle power conversion system.

PEACER is a Pb-Bi-cooled fast reactor being developed at the Seoul National University, designed for power production and waste transmutation. PEACER incorporates a pancake-type core with a U-Pu-Zr metallic fuel with a high thermal conductivity in square lattice cooled by forced circulation by a main coolant pump (MCP), and the Rankine cycle for power generation. As with other Pb-Bi cooled fast reactor concepts, the operating coolant temperature is low, spanning 300 ~ 400 °C to achieve corrosion-resistant conditions and a longer reactor lifetime.

PEACER provides two reactor designs of different capacity. PEACER-550 has a 1560 MWth core, following the basic integral fast reactor design. PEACER-300 is designed to produce 850 MWth. There is no intermediate heat transport system.

The steam at the turbine inlet is superheated to 633.15 K and 8 MPa. The thermal efficiency is estimated to be 35.3%.

PEACER is equipped with an active reactivity control and shutdown system (motor driven) and a passive reactor shutdown system (gravity driven). The active reactivity control and shutdown system consists of twenty-eight control assemblies that are used for power control, burn-up compensation and reactor shutdown.

BORIS is being developed as a multipurpose integral optimized fast reactor with an ultra-long-life core at the Seoul National University. BORIS aims at satisfying various energy demands, maintain inherent safety using the Pb coolant, and improved plant economics. BORIS is being designed to generate 22.2 MWth with 10 MWe for at least twenty consecutive years without refuelling and to meet the Generation IV nuclear energy system goals of sustainability, safety, reliability, and economics. BORIS is conceptualized to be used as the main power and heat source for remote islands and barren lands, and also considered to be deployed for desalinization purpose. BORIS consists of modular components to enable rapid construction and easy maintenance, and incorporates an integrated heat exchanger system operated by natural circulation of Pb without pumps to realize a compact reactor.

1.e The LFR and ADS designs considered in the handbook

The renewed interest in lead technology for critical fast reactors and ADS systems has resulted in the initiation of several projects, all at preliminary stage, most of them having been briefly described above. Their level of development, and their characteristics and objectives are in general very different and their prospects for full scale development are uncertain at the moment of the issuance of this handbook.

In the following discussion, details will be discussed for four systems which are more developed or present a better characterization of the potential of lead coolant technology for critical and subcritical systems.

The four systems selected for further discussion are SSTAR, ELSY, MYRRHA and EFIT.

1.e (i) SSTAR

The U.S. LFR Program is focused on the development of a small transportable reactor system known as the Small Secure Transportable Autonomous Reactor (SSTAR) with the following objectives [4]:

- Sealed core with no on-site refuelling or whole-core cassette refueling
- Transportability: the entire core and reactor vessel are delivered by ship or overland transport

- Long-life Core: 15-30 year core life is the target
- Autonomous load following with simple integrated controls: minimum operator intervention or maintenance required
- Local and remote observability: rapid detection/response to perturbations
- Minimum industrial infrastructure required in host location
- Very small operational (and security) footprint

In furtherance of the above objectives, current system development activities are being directed toward a pre-conceptual design and viability assessment for a SSTAR 20 MWe (45 MWth) natural circulation LFR for international deployment consistent with overall programmatic goals.

In addition, following the development of initial pre-conceptual designs, the LFR program was realigned to focus upon a concept for a near-term technology pilot plant to demonstrate successful reactor operation with a lead coolant at realistic system temperatures and incorporating innovative engineering that will help show the economic benefits and industrial attractiveness of Pb as a primary coolant.

A sketch of the current reference concept for the SSTAR small, modular, fast reactor is shown in Figure 5.2 [5]. This pre-conceptual design is a small shippable reactor (12 m X 3.2 m vessel), with a 30-year open-lattice cassette core and large-diameter (2.5 cm) fuel pins held by spacer grids welded to control rod guide tubes. The design integrates three major features: primary cooling by natural circulation heat transport; lead (Pb) as the coolant; and transuranic nitride fuel in a pool vessel configuration. 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 without 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 within which costs that are higher than those for large-scale nuclear power plants can still be considered competitive. Design features of the reference SSTAR in addition to the lead coolant, 30-year cassette core and natural circulation cooling, include autonomous load following without control rod motion, and use of a supercritical CO_2 ($S\text{-}CO_2$) Brayton cycle energy conversion system. The incorporation of inherent thermo-structural feedbacks imparts walk-away passive safety, while the long-life cartridge core life imparts strong proliferation resistance. If these technical innovations can be realized, the LFR will provide a unique and attractive nuclear energy system that meets Generation IV goals.

Some of the key design parameters of SSTAR are summarized in Table 5.1.

Table 5.1: Key Parameters of SSTAR

Power (MWe)	19,8
Conversion Ratio	~1
Thermal efficiency (%)	44
Primary coolant	Lead
Primary coolant circulation (at power)	Natural

Primary coolant circulation for DHR	Natural
Core inlet temperature (°C)	420
Core outlet temp. (°C)	567
Fuel	Nitride
Fuel cladding material	Si-Enhanced F/M Stainless Steel
Peak cladding temperature (°C)	650
Fuel pin diameter (mm)	25
Active core Height/diameter (m)	0.976/1.22
Primary pumps	none
Working fluid	Supercritical CO_2 at 20MPa, 552°C
Primary/secondary heat transfer system	N°4 Pb-to- CO_2 HXs
Safety grade DHR	Reactor Vessel Air Cooling System + Multiple Direct Reactor Cooling Systems

The research priorities of the SSTAR program are organized to address system design and evaluation, fuel cycle, energy conversion and material issues.

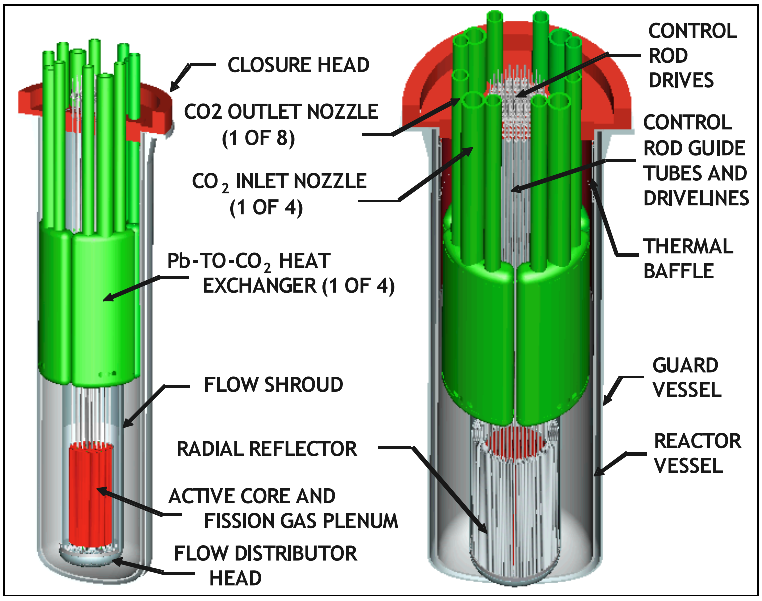

Fig. 5.2 - Conceptual 20 MWe (45 MWth) SSTAR system.

The R&D efforts required to advance the SSTAR concept are intended to address viability issues associated with the small transportable LFR and activities leading to the design and construction of a demo or pilot LFR plant. Viability will be established through focused R&D tasks in the areas outlined below and guided by formulation of a technically defensible pre-conceptual design.

- **System Design and Evaluation.** R&D tasks for System Design and Evaluation address the areas of core neutronics, system thermal hydraulics, mechanical design, passive safety evaluation, containment and building structures, in-service inspection, and assessing cost impacts. Core design is essential to establishing the necessary features of a 15 to 30-year-life core and determining core parameters that impact feedback coefficients. R&D tasks associated with this work include further optimization of the core configuration, establishing a start-up/shutdown rod and control rod strategy, and calculating reactivity feedback coefficients.
- **Fuel and Fuel Cycle.** Viability of both nitride fuel and whole-core cassette refuelling are to be addressed in the fuel and fuel-cycle R&D.
- **Energy Conversion.** Use of a S-CO2 Brayton cycle for energy conversion offers the prospect of higher thermal efficiencies with lower Pb coolant outlet temperatures and small turbo-machinery reducing the footprint and cost of the power converter.
- **Materials.** Viability of long core lifetime, passive safety, and economic performance (both capital and operating costs) of the LFR concept will depend on identifying materials with the potential to meet service requirements.

1.e (ii) ELSY

ELSY – the European Lead-cooled System – is a pool-type lead-cooled 600 MWe fast reactor, developed since September 2006, within the Sixth EURATOM Framework Programme (6FP).

ELSY aims at demonstrating the possibility of designing 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.

ELSY is an innovative project intended to globally address several of the most important technical challenges related to the use of lead technology in general, issues that have for the most part been only partially addressed in previous projects, namely:

- How to extend the LBE experience with LBE to pure lead as a coolant?
- How to mitigate the seismic issue?
- How to design a highly compact primary system?
- How to avoid in-vessel storage of spent fuel?
- How to cool high power spent fuel elements during refueling?
- How to design a compact SG?
- How to avoid the risk of catastrophic primary system pressurization associated with water or steam collector failure?
- How to mitigate the effect of a steam generator tube rupture (SGTR)?
- How to make the reactor internals removable?

- How to handle fuel elements while maintaining a temperature of 400 °C in lead?
- How to support the fuel elements in lead?
- How to design a simple and reliable safety-related decay heat removal (DHR) system?

The elimination of an intermediate cooling system and the development of a compact and simple primary circuit with all internal components removable, are among the features needed to assure reduced capital cost and construction time, competitive electric energy generation and long-term investment protection.

The relatively small size of the reactor vessel results from advanced solutions adopted for the primary system which features a cylindrical inner vessel, primary pumps installed in the inner zone of innovative flat-spiral-tube steam generators, and fuel elements substantially supported by buoyancy. In addition, the heads of the fuel elements extend above the vessel fixed roof as they are provided with long stems to allow fuel handling from the above reactor hall under full visibility.

In spite of the reduced coolant speed and of the moderate power density core, the innovative solutions adopted for ELSY allow reduced primary system dimensions (main vessel preliminary dimensions of 12.5 m diameter and 8.7 m height) which are similar to or even smaller than those of advanced pool-type SFRs.

Safety relies on the beneficial physical characteristics of lead, redundant and diverse DHR systems and other innovative features which make the primary system more tolerant to the effects of a SGTR accident.

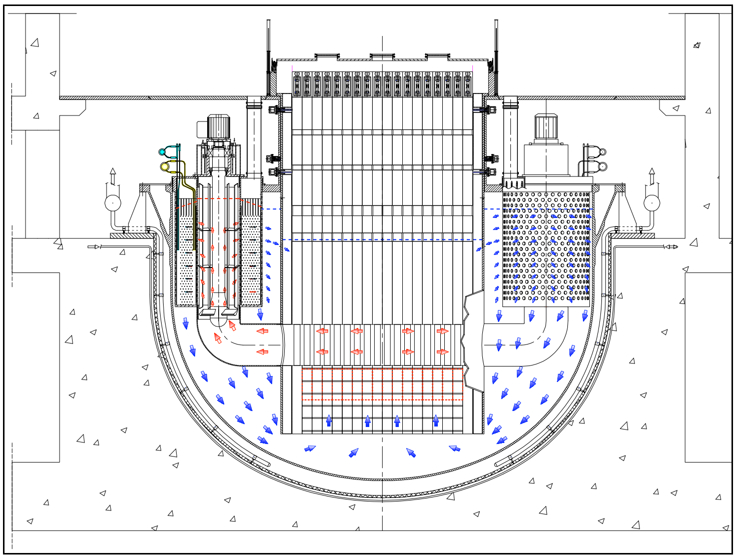

Fig. 5.3 - ELSY primary system arrangement and coolant flow path.

Table 5.2 provides the preliminary parameters of ELSY.

TABLE 5.2 - Main parameters of the ELSY plant

Power	600 MWe
Thermal efficiency	40%
Primary coolant	Pure lead
Primary system	Pool type, compact
Primary coolant circulation	Forced, at power, natural circulation + Pony motors for DHR
Primary pressure loss	~ 1.5 bar
Core inlet temperature	~ 400 °C
Core outlet temperature	~ 480 °C
Fuel	MOX with consideration also of nitrides and dispersed minor actinides
Fuel cladding material	T91 (aluminized)
Fuel cladding temperature	(max) ~ 550 °C
Main vessel	Austenitic stainless steel, hung, short-height ~ 9 m; diameter ~ 12.5 m
Safety vessel	Anchored to the reactor pit
Steam generators	N° 8, integrated in the main vessel
Secondary cycle	Water-superheated steam at 180 bar, 450 °C
Primary pumps	N° 8 mechanical, integrated in the steam generators
Internals	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	N° 4, DRC loops + a Reactor Vessel Air Cooling System .
Seismic design	2D isolators supporting the reactor building

1.e (iii) MYRRHA

Following the conceptual design phase of an experimental ADS conducted during the Euratom Framework Programme 5 (FP5) project PDS-XADS, a more advanced design for an Experimental Transmutation Accelerator Driven System, namely MYRRHA/XT-ADS, is being carried out within the FP6 Integrated project (IP) known as EUROTRANS and is proposed to be continued within the FP7 Central Design Team (CDT) in the near term period (*i.e.* through 2012). The major technological issues identified in this work are:

- System and plant design;
- Necessary dedicated R&D support issues;
- Material qualification programme;
- Fuel qualification programme
- High intensity proton accelerator performances and reliability

For the medium-term (*i.e.* to 2020), the emphasis will be on the construction of

MYRRHA/XT-ADS at Mol, Belgium. For the longer term, the development and qualification of innovative fuels (especially minor actinide-bearing inert fuels) with appropriate cladding and associated reprocessing techniques is a challenging task. Having these innovative fuels is mandatory to prove the technological feasibility of transmutation. Since the development of these innovative fuels will require a long time, research on this topic has already been started, but for the viability demonstration of ADS, it is of high importance to focus on current fuel qualification efforts of the driver fuel for fast spectrum systems.

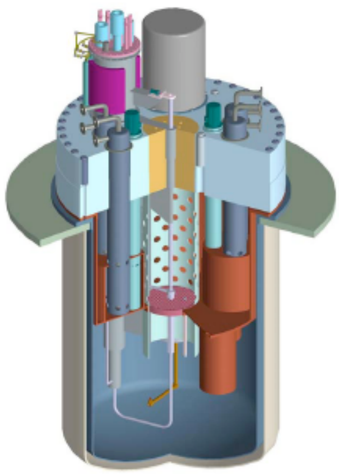

Fig. 5.4 - MYRRHA/XT-ADS primary system arrangement.

1.e (iv) EFIT

EFIT is the conceptual design of an European Facility for Industrial Transmutation (EFIT) with a pure lead-cooled subcritical reactor of about 400 MWth with the capability for minor actinide (MA) burning and electricity generation.

EFIT will be loaded with U-free fuel containing MA, namely (Pu, Am, Cm)O_{2-x} – MgO type fuel.

The neutronic design has confirmed the potential of EFIT to be an effective burner of MA with a net balance of -40.17 kg of MA/TWh and nearly a zero Pu balance (-1.74 kg/TWh).

MA burning, in addition to electricity generation, is an important added value of EFIT in the economic balance. However, a fraction of the electric power is used to produce the 16 MW proton beam and the accelerator and spallation target represent a significant part of the capital cost. It should be noted that the primary system volume per unit power is at least twice that of a pool-type SFR or LFR. Additional studies are necessary to understand if this penalty in the primary system dimensions is an unavoidable consequence of a subcritical system or can be reduced by optimization.

Based on the feedback from the operation of MYRRHA/XT-ADS and further progress on system design and fuel and material research, the construction of a European Facility for Industrial Transmutation (EFIT) can be envisaged as the final goal.

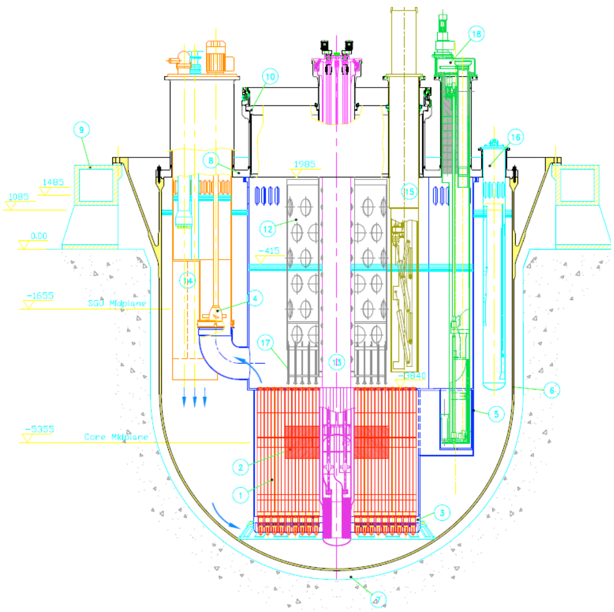

Fig. 5.5 - EFIT primary system arrangement.

2. Design Criteria and General Specifications

The requirements for the design of the LFR stem from engineering knowledge enhanced by the experience gained and the lessons learned in the field of sodium-cooled fast reactors (e.g. SPX1), in the Lead-Bismuth LBE technology for use in Russian submarines and in the technological limits acquired in the frame of several international experimental activities.

These requirements aim at achieving the main design missions of the LFR, which are the demonstration of its technical feasibility for electric energy generation and the demonstration of its capability to comply with the Generation IV goals (especially the capability of consuming Minor Actinides and of good economic performance).

Most requirements, such as the MA burning capability, are thus not absolute, however, and may undergo adjustment for optimization in the course of future design activities. It will be noted that the physical and neutronic properties of lead cannot be fully exploited simultaneously from the very beginning of the LFR design, because of technological and time constraints. The potential of burning MA from reprocessed spent fuel of LWR implies deployment of special, novel-design cores, which require qualification and testing in existing reactor systems.

On a global basis. priority is given to the demonstration of the technical feasibility of the LFR within a relatively short time frame with confirmation of structural material properties with effective oxygen control and with features such as a MOX-fuel core that is self-sustaining in Pu while being adiabatic in terms of consumption of the self-generated MA. It is expected that development of the LFR to the more ambitious goals of high temperature operation and burning capability beyond the self-generated MA will be pursued in the future and developed as appropriate depending on R&D and design achievements, and budget.

Compliance with the guidelines of Generation IV is an integral part of the LFR requirements. The tentative main design solutions are listed by the four Generation IV Goal Areas and the eight Generation IV Goals in the following sections. These solutions are deemed to be sufficient as starting features for a successful demonstration of the feasibility of the LFR.

The main LFR features identified in order to achieve the Generation IV goals are discussed below and summarized in Table 5.3. These features are based either on the properties of lead as a coolant or on specific engineered designs.

TABLE 5.3 - LFR potential performance against the four Goal Areas and the eight Goals for Generation IV

Generation IV Goal Areas	Goals for Generation IV Nuclear Energy Systems	Goals achievable via Properties of Lead	Goals achievable via Specific engineered solutions
Sustainability	Resource utilization	Lead is a low moderating medium.	Conversion ratio close to 1
	Waste minimization and management	Lead has low absorption cross-section. This enables a core with fast neutron spectrum even with a large coolant fraction	Great flexibility in fuel loading including homogeneously diluted MA
Economics	Life cycle cost	Lead does not react with water	Reactor pool configuration
		Lead does not burn in air	No intermediate coolant loops
		Lead has a very low vapour pressure	Compact primary system
		Lead is cheap	Simple design of the reactor internals
			Superheated steam or supercritical CO2 (high efficiency)
	Risk to capital		Small reactor size

	(Investment protection)		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;	Large fuel pin pitch Natural circulation cooling (small system) Decay Heat Removal (DHR) in natural circulation Primary pumps in the hot collector (moderate- or large- size system) DHR dip coolers in the cold collector
	No need for offsite emergency response	Lead density is close to that of fuel (considerably reduced 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.	Simplicity in design Independent, redundant and diversified DHR loops No use of reactive or flammable coolant materials

2.a Sustainability

According to Generation IV, *Sustainability* is the ability to meet the needs of the present generation while enhancing the ability of future generations to meet society's needs indefinitely into the future. Appropriate resource utilization and waste minimization and management are the two main aspects of the sustainability.

2.a (i) Resource utilization

Because lead is a coolant with very low neutron absorption and energy moderation properties, it is possible to maintain a fast neutron flux even with large amount of coolant in the core. This allows the efficient use of a variety of fuel materials, including fuels with homogenously mixed minor actinides. The reactor can be designed to achieve a Conversion Ratio of ~1 (without the need for a blanket), along with long core life and a high fuel burn-up. Preliminary results of the ELSY project indicate that a core with an active length of 0.9 m containing about 36 t of Heavy Metals (HM) is critical with 16.5% Pu and has a breeding ratio of about 1. Obviously a core with the same fuel content with a longer active length or larger fuel to coolant ratio will result in a breeding core.

The greatest experience on the use of heavy coolant is related to Lead-Bismuth Eutectic (LBE) because of its use in Russian submarine reactors and because of the technological development for sub-critical (ADS) reactors.

LBE is not considered, however, to be a sustainable coolant technology, because of the limited availability of bismuth. It is not proven that current bismuth resources will allow a large international deployment of central-station reactors, and its high cost, even at the present level, may represent as much as 10% of the plant capital cost. Lead is much more abundant than bismuth and much less expensive, and can be considered to be always available, even in the case of deployment of a large number of LFRs.

2.a (ii) Waste minimization and management

A fast neutron flux significantly reduces waste generation, Pu recycling in a closed cycle being the first condition recognized by Generation 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 and meet another important Generation IV condition.

Preliminary results of the ELSY core indicate the possibility to reach a MA content at equilibrium of 410 kg which corresponds to 1.1% of the fuel inventory.

Obviously loading the core with a MA content greater/lower than 410 kg will result in net MA burning/generating respectively.

2.b Economics

According to GEN IV, the economic goals broadly consider competitive life-cycle and energy production costs, reducing financial risks of nuclear energy systems. Additional use of nuclear energy is also considered, like low-temperature heat for water desalination or district heating and high temperature heat for hydrogen production.

The cost advantages of the LFR are expected to result from relatively low capital cost, short construction duration and low fuel production cost.

The economic utilization of Mixed Oxide Fuel (MOX) in a fast spectrum has been already successfully demonstrated in the case of the Sodium Fast Reactor (SFR), and a similar conclusion can be expected for the LFR.

Because of the favourable characteristics of molten lead, it will be possible to significantly simplify the LFR systems, and hence to reduce its overnight capital cost, which is a major cost factor for the competitive generation of nuclear electrical energy.

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 energy conversion equipment and the consequent elimination of the need for an intermediate system is a key-factor to provide competitive generation of electrical energy in the LFR. In the case of conventional steam cycle power conversion, this approach is possible because of the absence of fast chemical reactions between lead and water, although the SG tube rupture accident (*i.e.* risk of important pressure waves inside the SGU) must be considered in the design.

In the case of small-size plants such as SSTAR the use of molten Pb to CO_2 heat exchangers with supercritical carbon dioxide Brayton cycle energy conversion system can also be envisaged.

2.b (i) Risk to capital

The use of a new technology represents a potential risk for investors. Such risk must be overcome by innovative design features that bring about plant simplification and assurance of excellent economic performance.

Corrosion by molten lead of candidate structural steels for the primary system is a main issue in the design of a LFR. New materials are being sought for special components such as pump impellers. For near-term deployment, the use of classical materials for most of the reactor components will be made possible by limiting the core outlet temperature. In spite of this limitation, the overall system efficiency remains high because there will be no intermediate system to degrade the thermal cycle.

In-lead refueling and In-Service Inspection and Repair (ISI&R) of the core support structures in lead are additional critical aspects of licensing and operation.

In the ELSY project, it is proposed to face these apparent drawbacks by reducing the number of components/machine operating in lead, in particular by eliminating the core support plate and the in-vessel fuel transfer machine, which has, as yet, never been designed or tested in lead.

This is considered possible because the very low vapour pressure of molten lead should allow relaxation of the otherwise stringent requirements for gas-tightness of the reactor head and this allows the adoption of simpler solution.

In general, for small, transportable systems, a limitation to the risk to capital results from the small reactor size, With particular relevance to the central station system, a reduction in the risk to capital results from combining plant simplification with the design of removable/replaceable in-vessel components.

2.b (ii) Other use of nuclear heat

In a future expanded market of nuclear energy it is expected that additional uses of nuclear energy will be sought. For example, low temperature heat for water desalination or district heating can be readily envisioned. In this respect, the LFR can play a role similar to other nuclear power reactors and, in particular, it will favour modular applications. In the case of large hydrogen demand, the LFR could provide electricity for hydrogen generation by water electrolysis. The high boiling temperature of lead is potentially exploitable for hydrogen generation by high temperature chemical processes, but this possibility is conditioned to time-consuming development/use of new materials that are resistant to lead corrosion at higher temperatures.

2.c Safety and Reliability

Pure lead as a coolant is chemically inert in comparison to sodium and, moreover, it is preferred to LBE in term of safety because of its substantially lower radiological concern, especially Polonium-210.

One of the advantages in the use of lead in a fast reactor is the fact that lead retains hazardous radionuclides like iodine and cesium, even in the event of a very severe accident involving the failure of the reactor vessel, failure of the reactor building and exposure of the coolant to the atmosphere. This advantage is considerably reduced in the case of LBE because of the much higher production of the radiotoxic polonium-210. The polonium production in an LBE-cooled reactor is so high that, in the 80 MW LBE-cooled ADS developed in the 5th Framework Programme of Euratom, the polonium inventory was evaluated to be 2 kg (8.9 MCi) at equilibrium. This amount of polonium produces a decay heat in the primary system that equals the fuel decay power, after five days of cooling.

Pure lead is not exempt from polonium formation. In pure lead, ^{209}Bi is produced from ^{208}Pb, and ^{210}Po results from the further activation of ^{209}Bi; however, the rate of Po production is less by about 4 orders of magnitude than in case of LBE, and its decay power is negligible in comparison to that of the fuel.

The slightly higher density (4%) of lead in comparison to LBE has a marginal negative impact on the mechanical design, but the fact that the lead density is higher than that of an oxide fuel is beneficial in the event of an hypothetical severe accident, because the high density can further reduce the risk of re-criticality following a core melt.

In-vessel fuel handling is facilitated by the use of LBE which allows operation at lower temperatures, but the ex-vessel fuel and component handling is facilitated by pure lead because of the reduced polonium inventory.

In general, pure lead's characteristics facilitate the fulfilment of Generation IV objectives.

2.c (i) Operation will excel in safety and reliability

Safety is based both on the properties of lead as well as on the engineered solutions mentioned in the specific projects to meet the safety objectives. 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 vapour pressure, 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 gas, and the reactor operation safely resumed. Any leaked lead would solidify without significant chemical reaction affecting the operation or performance of surrounding equipments or structures.

Due to the low moderating capability of lead it is possible to have relatively large spacing among the fuel rods with low pressure losses in spite of the high density of lead. In ELSY and EFIT, the specified moderate core ΔT between the inlet and outlet temperatures not only minimizes the potential for material corrosion but also the thermal stress during transients, and the relatively low core outlet temperature minimizes the creep in steels.

In ELSY, an innovative reactor layout such as primary pumps installed in the hot collector, has been developed which, besides the economic advantages, improves several safety-related aspects, such as:

- Moderate volume of hot collector and large volume of cold collector.
- DHR Coolers immersed in the cold collector. This favors natural circulation and eliminates the interference between hot coolant streaming from the core and cold coolant from the outlet of the DHR Coolers.
- Free-level of the cold collector, in normal operation, higher than the free-level of the hot collector. This, in case of primary pump shutdown, favors a mild transition from forced to natural circulation of the coolant and hence ensures adequate heat removal from the core during a transient.

The installation of SGs inside the vessel is the real safety challenge of a LFR design.

Preventing and mitigating provisions must be conceived to address the possibility of high pressure water and steam release into lead. These measures will be directed toward reducing the frequency of occurrence of such releases, reducing the potential amount or rates of such releases, and mitigating the consequences.

A robust SG design and an appropriate plant operation and ISI program is necessary to reduce the frequency of release.

A typical provision to reduce large releases is the elimination of the risk of failure of the water and steam collectors inside the primary boundary by installing them outside the reactor vessel. This provision aims to eliminate by design a potential initiator of a severe accident of low probability, but potentially catastrophic consequences. The associated accident scenario has never been evaluated, but the complete disruption of the primary boundary and even of the overall core cannot be excluded.

In the case of SG tube rupture, a sensitive and reliable leak detection system coupled with a fast steam generator depressurization and isolation system are the basic features to minimize the risk of damage. High reliability requires redundancy of the leak detection system achieved by means of (i) acoustic system, (ii) steam detection in the reactor cover gas, and (iii) pressure increase detection of the reactor cover gas. Fast depressurization from high pressure in a few seconds will be achieved by operating on both the water side as well as steam side.

Several provisions can be provided to mitigate the consequences of the SG tube rupture (SGTR) accident which typically are the modification of the primary system chemistry, the pressure wave formation and propagation inside the primary system and the pressurization of the primary boundary.

To this end, three provisions have been conceived in ELSY:

- The first provision is the installation on each tube of a check valve close to the steam header and of a Venturi nozzle or flow blockage device close to the feed water header. With these devices, reverse steam flow is prevented and any leaking tube is, at least partially, promptly isolated.
- The second provision aims at ensuring that the flow of any feedwater-steam-primary coolant mixture be re-directed upwards and the risk of potentially disruptive pressure surges within the reactor vessel prevented by design. To this purpose in the event of a SGTR the normal radial flow is deviated upwards by design features that are fully passive and are actuated by pressurization in the SG bundle. Thus, thin perforated companion inner and outer shells are placed in the annulus close to inner and outer shell respectively, held apart to a few mm by spacers. The spacers are designed to collapse in the event that the inner companion shells are acted upon by a specified inner pressure. Thus, in case of an inner pressure surge, the companion shells would blow out against the inner and outer shell (Figure 5.5) respectively and since the holes of the corresponding perforations have been designed staggered and the bottom end of the annulus is closed, there will be no other exit path for the mixture, but the upwards

flow towards the cover gas plenum, that damps the pressure surge without risk of serious damage of the reactor internals.

- As a third provision, pressure relieving ducts, each with two rupture discs, installed on top of each SG unit, hydraulically connect the reactor cover gas plenum with the above-reactor enclosure in case of inner pressure surge, particularly brought about by the SGTR accident.

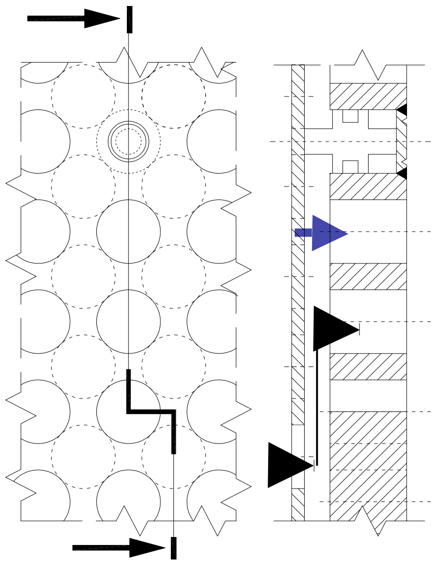

Fig. 5.6 - Double SG outer shell.

2.c (ii) Low likelihood and degree of core damage

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 control and protection systems.

In case of leakage of the reactor vessel, lead is collected inside a safety vessel and the coolant is designed to maintain a minimum level that ensures the coolant circulation through, and the safe heat removal from the core. In ELSY a specific solution has been developed with spiral-tube SG feed from the bottom which, without any penalty on the main vessel height, maintain a lead flow path beneath the minimum level.

For small-size reactors, since the vessel outer surface is relatively large in comparison with the reactor power, Decay Heat Removal (DHR), can simply be accomplished by a Reactor Vessel air Cooling System (RVACS) which is a very reliable system. For medium and large size reactors additional safety grade systems are needed. The fact that molten lead does not react violently with air or water gives the designer some freedom in the choice of the liquid for the DHR loops, the use of air and water remaining the preferred and most simple approach.

For power control and reactor shut down two completely different strategies are applied in case of ADS system or critical LFRs. In the case of an ADS, power level is controlled by means of the generated proton beam current.

In case of a critical LFR diversified solutions are possible, in general, based on the control rod technology similar to SFRs, even if the use of lead as a coolant increases the spectra of the potential solutions. A control/shutdown rod can replace a fuel element in the core layout, or can be located inside a fuel element. A control rod can be moved by electrical equipment located in the gas space. A shutdown rod can be introduced from the bottom of the core by lead buoyancy, from the top through motor-driven action, by the gravity of structural masses located in gas space, or by gravity-driven action inside an evacuated tube.

At the date of issuance of this document, several solutions/proposals are under investigation, but with large uncertainties and only after an appropriate test campaign in lead it will be possible to select the most promising solutions and confirm the level of reliability and diversification that can be achieved.

2.c (iii) Reduced need for offsite emergency response

In the LFR, fuel dispersion dominates over fuel compaction, thus reducing considerably the likelihood of the occurrence of severe re-criticality events in the case of core disruption. In fact lead density, which is slightly higher than the that of the fuel, and convective streams make it rather difficult to achieve scenarios leading to fuel aggregation with subsequent formation of a secondary critical mass, in the event of postulated fuel failure.

In addition the ability of lead to trap and retain fission products, in particular iodine and cesium, and the fact that a loss of coolant accident (LOCA) will not result in significant pressurization of the containment are features of primary importance in reducing the potential consequences of severe accidents.

2.d Proliferation Resistance and Physical Protection

The physical characteristics of lead, the selected fuel cycle and the adopted design features can contribute to increase the Proliferation Resistance and Physical Protection (PR&PP) characteristics of a LFR. For PR nevertheless international safeguards for each of the major elements of the system fuel cycle remain an independent assurance against potential diversion of nuclear fuel to produce or provide materials for nuclear weapons.

2.d (i) Unattractive route for diversion of weapon-usable material

The use of MOX fuel containing MA increases proliferation resistance (PR) because of the inherent properties of the nuclear material. Uranium enrichment is not necessary. A breeding (conversion) ratio close to 1 is achievable in a medium size reactor without providing fertile regions in the core and hence improving PR. Fertile regions can nevertheless be necessary to maintain a breeding ratio close to 1 in small reactors or to achieve a higher level of breeding.

Moreover, the SSTAR system has been designed from the beginning to achieve non-proliferation goals by incorporating a sealed core and very long life fuel.

High burn-up and hence a high spent fuel radiological barrier (up to about 100 GWd/tHM in the short term, up to about 200 GWd/tHM in the longer term when adequate structural material for fuel cladding has been made available) improves PR.

Other additional benefits which can result from the fuel cycle are the introduction of pyro- or advanced aqueous fuel reprocessing methods featuring incomplete removal of fission products and MA, no separation of uranium and plutonium at any fuel cycle stage, and inherently low decontamination factors for fuel, with the need for remote handling, which complicates operations but enhances proliferation resistance.

2.d (ii) Increased physical protection against acts of terrorism

The use of a coolant chemically compatible with air and water and operating at ambient pressure greatly enhances physical protection (PP). 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. There are no credible scenarios of significant containment pressurization. Significant PP features of the LFR systems include:

- system simplification, no intermediate cooling system, and consequent robustness;
- passive decay heat removal;
- compact security footprint;
- possibility of partial or full underground siting.

3. Neutronics

Fast reactors cooled by HLMs such as lead or LBE rely primarily on the physics of very high energy neutrons: the high mass number of lead (and bismuth) results in the maintenance of a very hard (high energy) neutron spectrum.

A typical neutron energy distribution in a LFR is shown in Figure 5.7 referring to ELSY. The mean neutron energy in a typical LFR lies in the range of 400 to 450 keV (depending also on the fuel type, *i.e.* oxide, nitride or metallic).

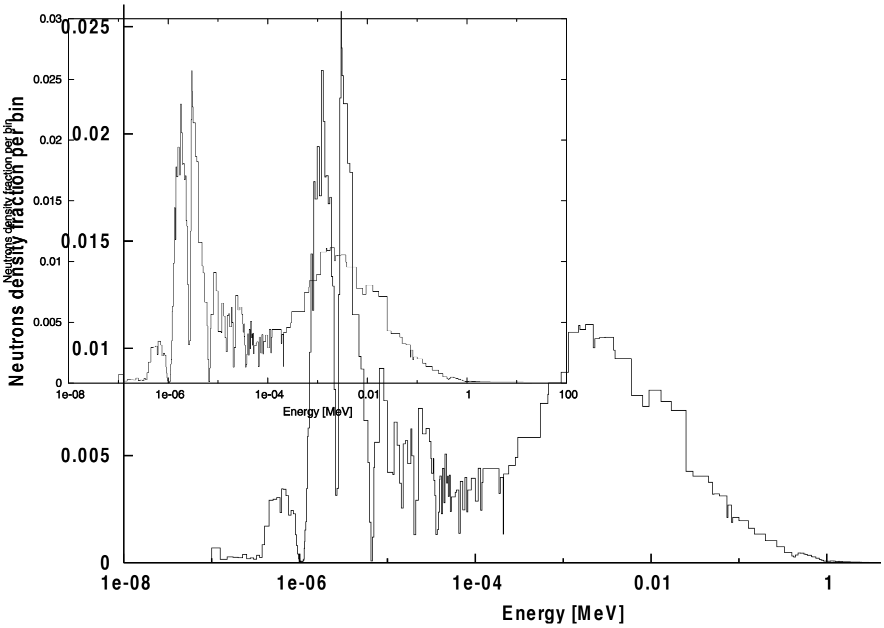

Fig. 5.7 - Typical neutrons spectrum in a LFR (e.g.: ELSY).

The mean free path associated to the given spectrum, for a typical LFR, is of the order of 2 to 3 cm.

3.a Neutronic properties of lead

In order to investigate the peculiarities of a LFR (for comparison with other reactor types) it is important to consider the range of neutronic properties of the coolant including moderation (slowing down) and absorption affinity.

3.a (i) Moderation

The hardness of the neutron energy spectrum described in the previous section and depicted in Figure 5.7 can be understood by taking into account the average lethargy change per elastic collision,

$$\xi = 1 - \frac{(A-1)^2}{2A} \ln\left(\frac{A+1}{A-1}\right) ,$$

where A is the mass number of the considered isotopes. Table 5.4 resumes typical values of the average lethargy change per elastic collision and the moderating power for lead and other main coolants/moderators.

TABLE 5.4 - Average lethargy change per elastic collision and moderating power for some typical coolants/moderators

	ξ	$\xi\Sigma_s$
H_2O	0.920	1.425
D_2O	0.509	0.177
Helium	0.425	9.0E-6
Graphite	0.158	0.083
Sodium	0.0825	0.0176
Lead	0.00963	0.00284

The elastic cross-section of naturally occurring lead isotopes is shown in Figure 5.8. In the energy range of interest the elastic cross section assumes almost the same value for all isotopes.

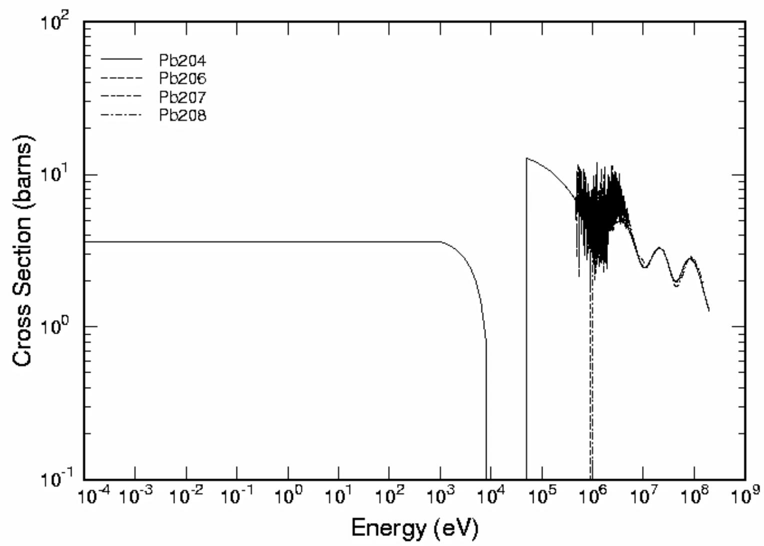

Fig. 5.8 - Elastic cross-sections of naturally occurring lead isotopes (data from ENDF/B-VII library).

3.a (ii) Absorption

The lead coolant is one of the main contributors to the neutron balance in the core: as a matter of fact, captures in the coolant directly impact the reactivity of the unit cell of the system, and thus the neutronic design of the whole core (see section 5.6).

The (n,γ) absorption cross-section of naturally occurring lead isotopes is shown in Figure 5.9.

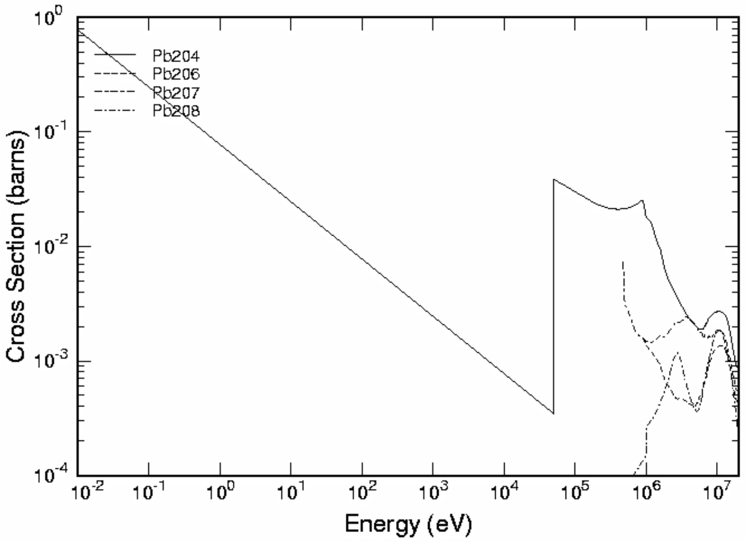

Fig. 5.9 - (n,γ) absorption cross-sections of naturally occurring lead isotopes (data from ENDF/B-VII library).

It is to be pointed out that the most highly absorbing isotope (^{204}Pb) has a natural abundance of only 1.4%, sustaining the low total absorption rate of lead.

3.b Fuel performances in LFRs

Whilst a variety of fuels is accounted worldwide for LFRs (e.g., oxide in the European concept, nitride in the American one), their typical composition is a mixture of reactor-grade Plutonium (referring to an isotopic vector as if extracted from the spent fuel of a typical LWR after a mean Burn-Up (BU) of some 50 GWd/tHM and a cooling period of 10 y) and Depleted Uranium (DU), eventually doped by the inclusion of some Minor Actinides (MAs). The only exception to this scheme is represented by the EFIT fuel: it is made of Pu oxide with a considerable fraction siderable fraction of MA-oxide only (thus U-free) in an inert matrix (MgO in the preliminary hypothesis).

Besides the peculiarities of oxide vs. metallic fuels (mainly influencing the thermal design of the pin, such as the fuel thermal conductivity and its melting temperature), a series of common properties can be pointed out referring to the overall performances of fissile and fertile isotopes in a LFR.

3.b (i) Fission cross-sections

An immediate drawback related to the hard neutron spectrum can be found in what concerns the fission cross-sections of odd nuclides (about one to two orders of magnitude less than in thermal reactors): typical values are shown in Table 5.5 compared to corresponding capture cross-sections. Despite the fact that an increase of the fission rates for even nuclides can be gained, resulting in a wider contribution to criticality among nuclides in the fuel inventory, the reduction of the fission cross-sections implies larger inventories of fissile material to maintain criticality.

TABLE 5.5 - Typical microscopic cross-sections of main fuel isotopes in a LFR compared to the ones of LWRs.

	Capture [barns]			Fission [barns]		
	ELSY	ENHS[1]	LWR	ELSY	ENHS	LWR
U238	0.282	0.210	1.03	0.035	0.030	0.107
Pu239	0.487	0.297	42.23	1.753	1.640	101.02
Pu241	0.475	0.313	37.89	2.501	2.110	109.17

3.b (ii) Average number of fission neutrons

The hard spectrum represents a positive contribution it what concerns the average number of neutrons per fission, $\bar{\nu}$, which is higher (about 2.93 for almost all the systems considered in the present chapter) than in thermal reactors. The higher number of neutrons available in the system, once criticality has been achieved, can be exploited for captures in fertile material to provide a higher breeding.

[1] The Encapsulated Nuclear Heat Source (ENHS) is a SSTAR-type reactor (see section 5.1.e (i)) candidate conceived by the University of California – Berkeley, the Lawrence Livermore National Laboratories and the Argonne National Laboratories.

3.b (iii) Fuel utilization

Supported also by the increase of $\bar{\nu}$, the fertility factor, η, increases monotonically above 100 keV: the main reason for this can be ascribed to the lower capture rate due to the higher separation of the bulk of the neutron spectrum from the absorption resonance energy range. LFRs therefore can rely on a more efficient fuel utilization, allowing a higher relative arrangement of fertile material in the fuel, thus resulting in a higher Conversion Ratio (CR).

3.b (iv) Spectrum evolution with Burn Up

The particularly hard spectrum of LFRs reduces the impact of the build up of Fission Products (FPs) on the neutron balance during operation (few tens of pcm after complete irradiation of the fuel). Hence the neutronic properties of the system can be assumed to remain approximately constant during the whole core life.

3.b (v) Effective delayed neutron fraction and prompt neutrons lifetime

In a typical LFR with iso-breeding Pu content (such as ELSY or SSTAR), the value of the effective delayed neutron fraction β_{eff} is in the range 370 (ELSY) to 420 (SSTAR) pcm. This value is smaller than that of LWRs (~ 650 pcm) because of the lower fraction of delayed neutrons per fission of a ^{239}Pu isotope than for ^{235}U. In case of MA-doped fuel (with equilibrium concentrations, *i.e.* some 1 at.% of HM) the value of β_{eff} is further reduced to some 325 pcm because of the small delayed neutron fraction associated to the fission of Minor Actinide isotopes.

The impact of more highly energetic neutrons also implies a lower prompt neutrons lifetime, λ, (of the order of 10^{-6} to 10^{-7} s) in comparison with thermal reactors (about two orders of magnitude higher).

The direct drawbacks related to the values of these parameters are the narrower margin to prompt-criticality and the lower capabilities for reactor control in case of prompt-criticality accident.

3.b (vi) LFR capabilities of MAs transmutation

Finally, the harder the spectrum, the higher the fission cross sections of MAs (triggering the highest level of threshold fission reactions among even nuclides). As far as MA transmutation is concerned, this implies that the balancing of production and removal rates for the latter (which represents the frontier between MA breeders and burners) is attained by a low content of MAs in the fuel.

The possibility of relying on a low fraction of MAs in the fuel allows more flexibility in waste transmutation for LFRs: performances being equal, the lower detriment to the total average fraction of delayed neutrons (since the low contribution associated to MAs) represents a larger operability margin to what concerns such a stringent constraint for reactor control.

3.c Neutronic performances of typical absorbers in a LFR

The choice of an effective neutron absorbing material is fundamental in the design of a critical reactor because of the need to control and regulate its operation. In the hard spectrum of a LFR. particular care should be paid to the evaluation of the absorption cross-sections of the control material candidates.

3.c (i) Boron Carbide

Boron carbide, B_4C, is the reference absorbing material for FRs in general. Besides the availability of boron and the ease of its fabrication, therefore, the low costs related to the adoption to B_4C based control systems, the neutronic properties of the ^{10}B isotope are excellent even in the fast spectrum because of its (n,α) reaction cross-section (Figure 5.10).

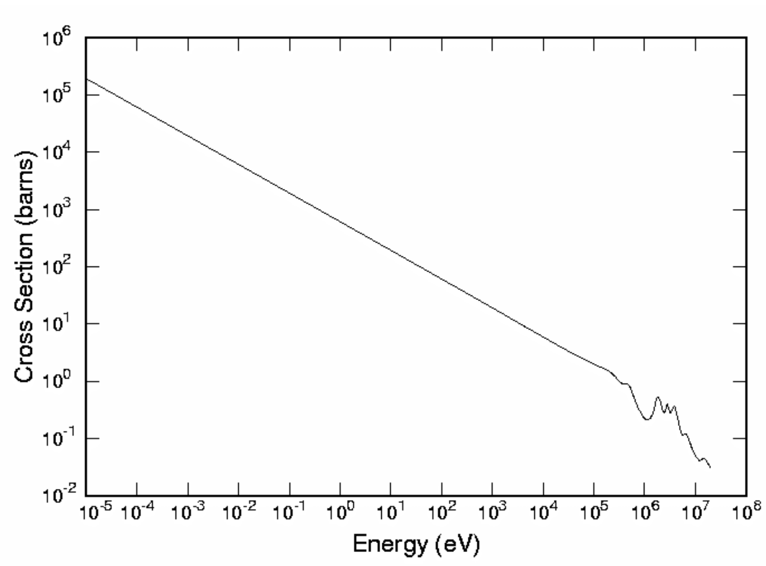

Fig. 5.10 - ^{10}B absorption cross-section (data from ENDF/B-VII library).

Since the main isotope responsible for neutron absorption is the ^{10}B isotope, the natural abundance of which is 19.9% (the rest being ^{11}B, whose neutronic properties are practically unusable), reactor grade B_4C is usually enriched in ^{10}B (up to ~ 90%).

3.c (ii) Indium-Cadmium eutectic

An interesting alternative to boron carbide could be represented by the indium-cadmium eutectic (75 wt.% In and 25 wt.% Cd): at typical LFR operating temperatures (> 400 °C), this alloy is liquid (T_{melt} = 122.5 °C); thus its operability could be assured even in case of Control Rod (CR) thimble guide distortion after a severe accident.

A main drawback can be ascribed to this solution taking into account the absorption effectiveness of the eutectic in the hard spectrum of LFRs. Despite the high content of indium in the mixture (the absorption effectiveness of pure cadmium being about 60% of that of pure indium), preliminary evaluations performed on an ELSY In-Cd control system, when replacing a reference 90% ^{10}B enriched B_4C configuration, showed reactivity reduction to about 14% of the latter configuration. Based on this assessment, large volumes in the core would need to be devoted to such a control system, resulting in the practical infeasibility for this solution.

3.c (iii) Europium

The last candidate absorber for LFRs is europium sesquioxide (Eu_2O_3). This material, well known in reactor physics, has high (n,γ) absorption cross-sections in the fast spectrum (comparable to that of ^{10}B for both naturally occurring isotopes ^{151}Eu and ^{153}Eu, Figure 5.11).

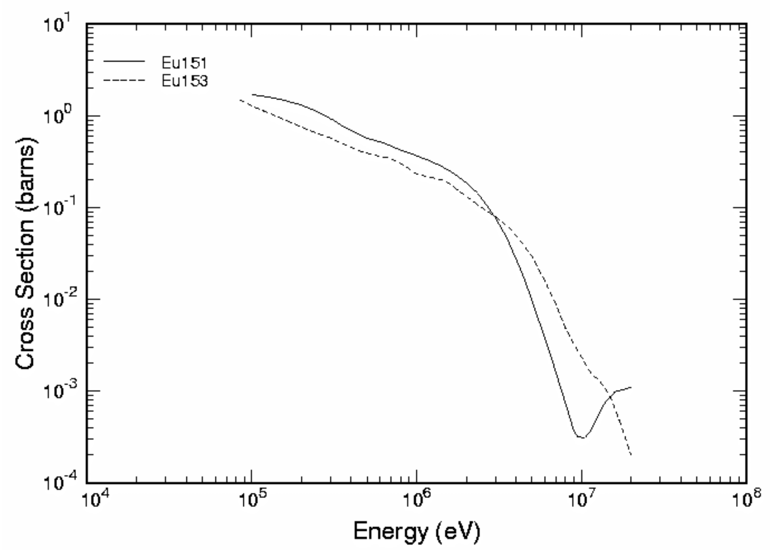

Fig. 5.11 - Naturally occurring europium isotopes (n,γ) absorption cross-section (data from ENDF/B-VII library).

In order to decide whether to choose this material instead of B_4C it is worth taking into account that

- no He is produced (since the capture mechanism is Eu radiative absorption), thus no venting is required for europium sesquioxide CRs;
- the daughter products are also good neutron absorbers, thus the loss of anti-reactivity worth is reduced with respect to B_4C absorbers;
- the Eu self-shielding is such that the effectiveness of a Eu_2O_3 CR, in LFRs, is about 40% of that of an equivalent one made of enriched B_4C, thus close to that of natural B_4C.

4. Lead Properties

This chapter reports data on the main physical properties of technically pure molten lead[2] with a few complementary data of LBE acquired from the open litera-

[2] Technically pure lead is not synonymous with nuclear grade lead, because lead as a coolant in a fast reactor is likely to require more stringent limitations than high-purity industrial lead, in term of concentrations of impurities, which could become activated or affect corrosion, mass transfer and scale formation on heat transfer surfaces. The impurity concentrations are so low, however, that the physical properties of lead of both grades are the same, for the purpose of this compilation.

ture. It will be noted that, in some cases, these data are recommended as best-fit data, because of the significant discrepancies among the values given by different sources.

4.a Physical Properties

The properties of molten lead are given by parameter in the form of the recommended value or correlation over a temperature range, and, eventually, in the form of a table of main parameters relevant to heat transfer, of both pure lead and the lead-bismuth eutectic over the range of more frequent use, for quick check and comparative analysis purpose.

In general, the reliability of the recommended correlations about thermal-physical property data of molten lead is satisfactory and the correlations can be used for engineering estimates and design calculations, in spite of the uncertainty still existing on heat capacity, boiling temperature and thermal conductivity.

For the high temperature range, the set of thermodynamic and transport properties (thermal conductivity, viscosity and surface tension) relevant for reactor safety analysis are not reported, but the basic properties such as the liquid density, vapour pressure and liquid adiabatic compressibility, are estimated up to the critical point using semi-empirical models based on the extrapolation of low temperature data [NEA ***NEED TO PUT IN REFERENCE TO NEA HANDBOOK IF THIS IS THE SOURCE OF THE FOLLOWING INFORMATION ***], owing to lack of experimental data published in the open literature.

<u>Normal melting point</u>

The melting point of technically pure lead is:

$$T_{melt}^{Pb}\ [K] = 600.6 \pm 0.1$$

The melting point increases by 0.0792 K per 1 MPa when pressure increases from about 15 to 200 MPa.

<u>Volume change at melting</u>

Similar to the majority of metals with FCC crystal structure, lead exhibits a volume increase upon melting. At normal conditions a volume increase

$$\Delta Vm/Vm = 3.7\%$$

is the recommended value for lead of technical purity.

<u>Latent heat of melting at the normal melting point</u>

The recommended heat of melting of lead at the normal melting point (the enthalpy change on melting) is

$$Q_{melt}^{Pb}\ [kJ\ kg^{-1}] = 23.8 \pm 0.7$$

Normal boiling point

The value of:

T_{boil}^{Pb} [K] = 2016 ± 10

is recommended for the boiling temperature of technically pure lead at normal conditions.

Heat of vaporisation at the normal boiling point

The latent heat (enthalpy) of vaporisation is a measure of the cohesive energy of atoms in a liquid metal. Therefore, it correlates with surface tension and thermal expansion. The literature values are very close, with the difference between maximum and minimum values less than 1%. The mean value and the mean deviation are:

Q_{boil}^{Pb} [kJ kg^{-1}] = 858.2 ± 1.9

Saturation vapour pressure

The vapour pressure of a liquid metal is an important property which is directly related to the latent heat of evaporation (cohesive energy, ΔH). The following correlation is recommended for the saturated vapour pressure of molten lead where temperature is in K:

p_s^{Pb} [Pa] = 6.189 *10^9 * exp(-22216 / T)

The above equation can provide approximate values for equilibrium vapour pressures over a wide range of temperature and is recommended from the melting point up to the normal boiling point. ΔH is included as a constant owing to the relatively small variation with temperature.

Surface tension

The surface tension of liquid surfaces (σ) is related to tendency to minimise the surface energy. It decreases with increasing temperature and reduces to zero at the critical temperature (T_c) where difference disappears between liquid and gas phases.

The temperature dependence of surface tension is linear for most liquid metals.

The recommended correlation is the following formula where temperature is in K and is conservative in the range from the melting temperature of 600.6 K to 1200 K (327.6 to 927 °C):

σ^{Pb} [N m^{-1}] = 0.519 - 1.13*10^{-4} * T

Density

The temperature dependence of density provides essential information for the development of an equation of state (EOS). It is used to determine the concentration of atoms in unit volume and hydraulic parameters in reactor design. Also, the measurement or calculation of basic physical properties, e.g., viscosity, surface tension, thermal diffusivity, requires knowledge of density. The set of the selected data can be fit as follows, with linear temperature dependence, where temperature is in K:

ρ^{Pb} [kg m^{-3}] = 11367 - 1.1944 * T

Thermal expansion

The coefficient of thermal expansion (CTE), derived from temperature dependence of density of molten lead is:

β_p^{Pb} [K^{-1}] = 1/(9516.9 - T)

Sound velocity and compressibility

The correlation recommended for the estimation of the sound velocity in the molten lead is:

u_{sound}^{Pb} [m s^{-1}] = 1951.75 - 0.3423 * T + 7.635*10^{-5} * T

where the temperature is in K.

At normal atmospheric pressure, the temperature dependence of the elastic modulus of molten lead can be described with the help of parabolic and linear functions as follows:

BS^{Pb} [Pa] = (42.15 - 1.652*10^{-2} * T + 3.273*10^{-6} * T^2) * 109

Heat capacity

Available experimental data on heat capacity of heavy liquid metals are few. The following correlation is recommended for the heat capacity of molten lead in the temperature range of T_{melt} to 1300 K (1027 °C), where temperature is in K:

c_p^{Pb} [J kg^{-1} K^{-1}] = 175.1 - 4.961*10^{-2} * T + 1.985*10^{-5} * T^2 - 2.099*10^{-9} * T^3 - 1.524*10^6 * T^{-2}

Critical constants

Critical parameters. The mean rounded values of two sources are recommended for the critical temperature, pressure and density of lead:

T_c^{Pb} = 4870 K (4597 °C),
p_c^{Pb} = 100 MPa,
ρ_c^{Pb} = 2490 kg/m^3.

Viscosity

The following empirical equation, obtained by fitting selected values into an Arrhenius type equation, is recommended to describe the temperature dependence of the dynamic viscosity of molten lead:

μ^{Pb} [Pa s] = $4.55*10^{-4}$ * exp(1069 / T)

where temperature is in K. This correlation is valid in the temperature range Tmelt to 1470 K (1197 °C).

Electric resistivity

The electrical resistivity of liquid lead, as of most liquid metals, increases linearly with temperature (in the temperature region of interest).The recommended empirical equation suitable for the calculation of the electrical resistivity is as follows:

r^{Pb} [Ω m] = $0.666*10^{-6} + 4.79*10^{-10}$ * T

which is valid in the temperature range of 601 ÷ 1273 K (328 ÷ 1000 °C). The deviation of the selected data from this correlation is less than 1%.

Thermal conductivity and thermal diffusivity

Experimental determination of thermal conductivity of liquid metals is difficult because of the problems related to convection and wetting. At present, few experimental data are available, sometimes presenting discrepancies between different sets of data. The high thermal conductivity of liquid metals is mainly due to free electrons. A simple theoretical relation exists for pure metals between electrical and thermal conductivities known as Wiedemann-Franz-Lorenz law. In an effort to find a physically reasonable compromise among the experimental data sets and taking into account the relation with the electrical conductivity, the following linear correlation is recommended for the thermal conductivity of molten lead:

λ^{Pb} [W m^{-1} K^{-1}] = 9.2 + 0.011 * T

where temperature is in K. This correlation is applicable in the temperature range of Tmelt-1300 K (1027 °C).

Thermal diffusivity a is defined as follows:

$a = \lambda / (\rho * c_p)$

So, it can be calculated using data for thermal conductivity, density and specific heat. Fitting the data with a linear function yields a correlation which can be of practical use for calculating the dimensionless Péclet and Prandt numbers.

The values of main parameters and transport properties at discrete temperatures in the range 400 °C to 600 °C for lead (and LBE for comparison with pure lead) in Table 5.5.

TABLE 5.5 - Values of physical & transport properties of molten lead (and of the Lead Bismuth Eutectic alloy, for comparison)

Fluid	Temperature, °C	Density, ρ kg m^{-3}	Specific heat, c_p J kg^{-1} K^{-1}	Dynamic viscosity, η 10^{-4} Pa s	Kinematic viscosity, ν 10^{-8} m^2 s^{-1}	Thermal conductivity, λ W $m^{-1}K^{-1}$	Thermal diffusivity, a 10^{-6} m^2 s^{-1}	10^{-2} Pr= ν /a	Thermal expansion, β 10^{-4} K^{-1}	Surface Tension, σ 10^{-3} N m^{-1}
Lead	400	10563	146.7	22.3	21.1	16.6	10.7	1.86	1.12	443
	425	10533	146.3	21.0	20.0	16.9	10.9	1.71	1.13	440
	450	10503	145.9	20.0	19.0	17.2	11.2	1.59	1.13	437
	475	10473	145.5	19.0	18.1	17.4	11.4	1.48	1.13	434
	500	10444	145.1	18.1	17.4	17.7	11.7	1.38	1.13	432
	525	10415	144.7	17.4	16.7	18.0	11.9	1.29	1.14	429
	550	10384	144.3	16.7	16.0	18.2	12.2	1.21	1.14	426
	575	10354	143.9	16.0	15.5	18.5	12.4	1.13	1.15	423
	600	10324	143.5	15.5	15.0	18.8	12.7	1.07	1.15	420
LBE	400	10205	144	15.1	14.8	13.0	8.3	1.67	1.29	393
	425	10172	143	14.5	14.3	13.4	8.6	1.56	1.29	391
	450	10139	143	14.0	13.8	13.7	8.9	1.47	1.30	389
	475	10473	143	13.5	13.4	14.0	9.2	1.38	1.30	388
	500	10073	142	13.1	13.0	14.3	9.4	1.30	1.30	386
	525	10040	142	12.7	12.6	14.6	9.7	1.23	1.31	384
	550	10006	141	12.3	12.3	14.9	10.0	1.17	1.31	383
	575	9973	141	12.0	12.0	15.2	10.3	1.11	1.32	381
	600	9940	141	11.7	11.8	15.5	10.5	1.06	1.32	379

4.b Chemistry Control and Monitoring Systems

The chemical properties data of solubility and diffusivity of oxygen and some metallic elements, e.g. Fe, Cr, and some oxides (e.g. iron oxides, chromium oxides, etc.) in the molten lead are of paramount importance for:

- preventing oxidation of the coolant;
- the assessment of the materials corrosion rate;

- the design and engineering of HLM purification systems, for the development of a corrosion protection strategy that is based on protective oxide layers on the structural materials;
- the source term assessment.

The accuracy of the formula that fits the solubility data of oxygen is not reported in the literature, although solubility is one of the key parameters of lead chemistry. On lack of accurate data, a larger margin against risk of reaching saturation will have to be specified, particularly at the cold temperature of the thermal cycle (one decade from saturation, say). Diffusivity of oxygen is of lesser concern, because, although the diffusion rate in the melt of atomic oxygen is slow, the coolant flow itself will provide for uniform concentration of the dissolved oxygen wherever turbulent flow prevails at a rate that can be thousand times the rate of diffusion through stagnant lead.

It will be noted, at the outset, that the aim of controlled dissolved oxygen in the melt is to protect the structural steels such as stainless steels and low-alloy steels from corrosion by means of an oxide barrier and that this technique is effective up to about 500 °C.

This implies the presence of dissolved oxygen in the melt in equilibrium with oxygen gas in the cover gas plenum above the melt.

4.b (i) The thermodynamical base

Oxygen gas dissolves in liquid metals in atomic form. The amount of dissolved oxygen is proportional to the square root of its partial pressure above the melt (relationship known as Sievert's law), provided that its concentration is less than 1 wt.% and oxide-forming elements are absent. This holds true for the oxygen concentration range in the pure lead melt of LFR, the upper limit of which is the saturation concentration with respect to lead oxide and the lower limit the saturation concentration with respect to magnetite. Thus iron is kept fully oxidised (as magnetite) and lead fully reduced (as metallic lead).

All elements less noble than iron, if present, are a fortiori completely oxidised and all elements more noble than lead, if present, are in their metallic form. Elements, which are in between iron and lead, are kept at low concentration either by specification of the grade of the original lead charge (arsenic, bismuth) or by prevention from leaching out of steel into the melt (nickel) and play therefore a small to negligible role in the economy of dissolved oxygen concentration at normal operation or during tests.

Oxygen concentration changes as a consequence of changing partial pressure above the melt according to Sievert's law and/or of changing temperature of the melt. The rate of change due to changes of the partial pressure may be slow if diffusion is the only driving force or relatively fast if turbulent motion is involved. The rate of change due to local cooling (heating up) of the melt, as it occurs in the steam generators or in the core depends on the formation (dissolution) rate of the metal oxides.

It will be noted that any change of oxygen concentration is related to the dissolved oxygen. The total oxygen may have remained constant, while the oxygen concentration has changed, as in the case of formation of metal oxide particles or dissolution of metal oxide particles in the bulk of the melt.

The physical-chemical principles of metal oxide formation, particularly the relationship between free energy and equilibrium constant, are treated in every handbook on thermodynamics [Ref.1, Ref.2, Ref.3 ***NEED TO INCLUDE REFERENCES ON THE LIST AND ASSIGN CORRECT NUMBERS***] and concisely summarized here below.

The general chemical equation of metal oxide formation with one mole of oxygen is the reference for the calculations. Thus for lead the equation becomes

$$2Pb + O_2 \rightarrow 2PbO \qquad (5.1a)$$

Lead oxide formation in presence of water vapour can be conveniently expressed combining eq. (5.1a) and the equation of water formation

$$2H_2 + O_2 = 2H_2O \qquad (5.1b)$$

as follows

$$Pb + H_2O \rightarrow PbO + H_2 \qquad (5.1c)$$

Because the Gibbs free energy change ΔG is the driving force of a chemical reaction, on the equilibrium position ΔG vanishes, *i.e.,* an equilibrium mixture of both products and reactants is obtained when the free energy change of the system between the initial condition and the final condition has become zero (position of minimum free energy towards which the system tends). If the initial condition is taken to refer to standard conditions, the calculation of the standard free energy change $\Delta G^{\circ}{}_{T}$ at any temperature T allows the calculation of the equilibrium constant K at that temperature according to eq. (5.2)

$$\Delta G_T^{\circ} = -RT \ln K \qquad (5.2)$$

where:

R=Gas constant, 8.314 $JK^{-1}mol^{-1}$

T= Temperature, K

If Pb and PbO involved in eq. (5.1a) are pure liquid and solid respectively, their concentrations remain constant allowing their active masses to be taken as 1, and the equilibrium constant may be written in terms of the partial pressure of oxygen only

$$K = \frac{1}{p_{O_2}} \tag{5.3a}$$

and by substitution in eq. (5.2)

$$\Delta G_T^{\circ} = RT \ln p_{O_2}$$

allowing the dissociation or equilibrium p_{O_2} values to be calculated. The value of p_{O_2}, which is the reciprocal of K, is the equilibrium pressure of oxygen below which the oxide will decompose (therefore also called dissociation pressure) and above which the metal will oxidise. Low dissociation pressure favours more ready formation of the metal oxide.

In the case of the water vapor formation, eq. (5.1b) the equilibrium constant is in terms of partial pressures of all involved substances

$$K_p = \frac{p^2{}_{H_2O}}{p^2{}_{H_2} * p_{O_2}} \tag{5.3b}$$

In the case of dissolution of a metal oxide in the melt, $M_xO_y = xM + yO$, the equilibrium constant is in terms of the molar concentration of the solved elements

$$K_c = [M]^x * [O]^y \tag{5.3c}$$

The same reasoning as for the dissociation pressure applies considering the reaction of lead oxide formation in presence of water vapor, eq. (5.1c): at any temperature T, equilibrium exists only at that value of the $\frac{p_{H_2}}{p_{H_2O}}$ partial pressures ratio that satisfies eq. (5.3b), once the value of the dissociation pressure has been substituted for p_{O_2}. If the pressure ratio is kept lower than the equilibrium ratio, all lead is in the oxide form and, *vice versa*, all lead is in metallic form (see also Table 5.6, where related equilibrium values for both pressures ratio and oxygen dissociation pressure can be read at 400 °C for lead oxide and magnetite).

At low partial pressure of the oxygen gas above the melt, the knowledge of the equilibrium constant of water vapor offers a means of controlling p_{O_2} according to eq. (5.3b): at a given p_{H_2O}, p_{O_2} varies with $\frac{1}{p^2{}_{H_2}}$, the inverse of the square partial pressure of the hydrogen gas. At low p_{O_2}, the measurement of the associated hydrogen partial pressure is easy.

The standard free energy is defined in terms of enthalpy and entropy as follows:

$$\Delta G_T^{\circ} = \Delta H_T^{\circ} - T\Delta S_T^{\circ} \tag{5.4}$$

where conventionally $\Delta G_T^{\circ} = \Delta G^{\circ}{}_{T,prod} - \Delta G^{\circ}{}_{T,react}$, *i.e.* change in the free energy referred to the moles of the reactants and products shown in eq. (5.1a), at 1 atm and at a stated temperature T, with the substances in the physical state normal under these conditions. As can be seen from eq. (5.4), $\Delta G^{\circ}{}_T$ is made up of an enthalpic and of an entropic term.

The enthalpic term may be calculated as follows if the enthalpy change for the reaction at another specified temperature, usually 298 K, and molar heat capacity data are available from data books:

$$\Delta H^{\circ}{}_T = \Delta H^{\circ}{}_{298K} + \int_{298}^{T} \Delta C_p dT \pm latent \cdot heats \tag{5.5}$$

The entropic term may be calculated with the same procedure as for the enthalpic term as follows:

$$\Delta S^{\circ}{}_T = \Delta S^{\circ}{}_{298K} + \int_{298}^{T} \frac{\Delta C_p}{T} dT \pm \frac{(latent \cdot heats)}{T_m} \tag{5.6}$$

Latent heats are subtracted if reactants transform.

4.b (ii) Thermodynamical data and diagrams

Heats of formation can be read from Table 3-201 of Ref.1[**NEED CORRECT REFERENCES 1, 2 AND 3***] , absolute entropies at 298 K from Table 68 of [2], heat capacities from Table 3-173 of [1] and Table 2.2 of [3].

The following solubility at saturation, [wt.%], vs T of oxygen, eq. (5.7), and of iron, eq. (5.8), in molten lead have been determined experimentally [4 ***CORRECT REFERENCE***]:

$$\log C_{[O],s} = 3.438 - \frac{5240}{T}$$ (5.7)[3]

[3] Temperature range 400 to 700 °C

$$\log C_{[Fe],s} = 2.1 - \frac{4380}{T} \qquad (5.8)^{4}$$

<u>Change in the std free energy</u>

The change in the std free energy ΔG° vs T for the formation of two moles of PbO and half a mole of magnetite, Fe_3O_4, calculated with the relationships written above and the thermodynamic data, is plotted on the following diagram of Figure 5.12 along with iso- $\frac{p_{H_2}}{p_{H_2O}}$ pressure ratio, and iso- p_{O_2} pressure lines of interest. Selected values at 400 °C and 480 °C are reported in Table 5.6.

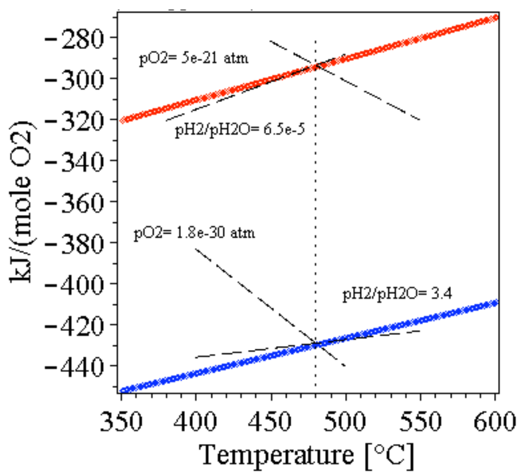

Fig. 5.12 - Std free energy change for the formation of two moles of PbO (upper red line) and half a mole of magnetite, Fe_3O_4.

The values of the dissociation pressures are so low that their direct measurement in the cover gas is impractical (Table 5.6).

TABLE 5.6 - Free energy change and dissociation pressures of lead oxide and magnetite at 400 °C and 480 °C

	$\Delta G^{\circ}_{400°C}$ kJ/mol O2	$p_{O_2\,400°C}$ atm	$\Delta G^{\circ}_{480°C}$ kJ/mol O2	$p_{O_2\,480°C}$ atm
PbO	-309.6	1.1*10-24	-293.5	5.0*10-21
Fe_3O_4	-442.8	4.5*10-35	-429.1	1.8*10-30

[4] Temperature range 400 to 900 °C

The oxygen activity is defined as the ratio, at any temperature, of the actual dissolved oxygen concentration to the concentration of saturation for PbO, arbitrary chosen as standard state:

$$a[O] = \frac{C_{[O]}}{C_{[O],s}} \qquad (5.10),$$

where $C_{[O],s}(T)$ is given by eq. (5.7).

Values of $C_{[O],s}(T)$ by step of 10 K are listed in the following Table 5.7. They are noted as standard values, because used as reference values in eq. (5.10) for calculating $a[O]$.

The activity a[O] gives the measure how far (how close) the actual oxygen concentration in the melt is from the upper allowable limit, where lead oxide would start to form. Because

$$C_{[O],a} = k\sqrt{p_{O_2}} \qquad (5.11)^{5}$$

the smaller $C_{[O],a}$, the smaller the $a[O]$ value until it corresponds to the dissociation pressure of magnetite via eq. (5.8) and eq. (5.11).

The theoretically allowable activity ranges are listed in Table 5.7 at several temperatures. The ranges cannot be exploited in their full extension, however, because if the oxygen activity would be controlled at is upper limit, lead oxide precipitation would occur in the melt, and predictably on the colder heat transfer surface of the SG tubes, while cooling down (too much dissolved oxygen) or, conversely, there would be dissolution of magnetite out of the mixed oxide film barrier of the austenitic steels, if the activity would be controlled at is lower limit (too little dissolved oxygen).

Thus, the operating activity range must be controlled between the two range limits with margin.

TABLE 5.7 - Wt.% oxygen dissolved in molten lead as function of the temperature at equilibrium with the dissociation pressures of lead oxide, [O]sat,std, and magnetite, [O]min

Temp, °C	Pure Lead		
	[O] sat, std wt.%	[O] min	
		wt.%	$-\log a[O]$
400	4.5e-5	2.94e-10	5.2

[5] Henry's constant k has the dimension [atm^{-1}], if the gas constant R = 1.98 cal/mol K.

450	1.6e-4	2.03e-9	4.9
480	3.0e-4	5.74e-9	4.7
500	4.6e-4	1.10e-8	4.6
550	1.2e-3	4.86e-8	4.4
600	2.7e-3	1.82e-7	4.2

4.c Thermalhydraulics

There are two kinds of open issues in this area. The first is related to the fundamental nature of heavy liquid metals.

The Prandtl number of HLMs (Pr of lead at 400 °C is 0.019) is more than 2 orders of magnitude lower than that of water and air. This is because HLMs have a significantly higher thermal conductivity λ [W/mK], lower specific heat capacity cp [J/(kgK)] and lower kinematic viscosity.

Low Pr number means that the thickness of the viscous boundary layer is negligibly small compared to the thermal boundary layer. In gas or water flow, the thickness of the thermal and the viscous boundary layer are of the same order of magnitude, as Pr is ~ 1.

With lead in laminar flow, molecular conduction of heat controls the heat transfer. Accordingly, the classical non-dimensional correlations for heat transfer can be applied also to liquid metal.

Under turbulent flow conditions, however, eddy conduction of heat becomes important and heat transfer is determined by both molecular and eddy conduction in the fluid stream. While in ordinary fluids like air and water molecular conduction is only of importance near the wall, in a liquid metal the magnitude of the molecular conductivity is of the same order as that of the eddy conductivity. Thus, the molecular conduction is effective not only in the boundary layer but also to a significant extend in the bulk of the fluid stream. Therefore, relationships (or correlations) developed to determine the heat transfer coefficients for turbulent flows in air or water cannot be used.

A further consequence of the importance of molecular conduction of thermal energy in turbulent liquid metal flow is that the concept of the hydraulic diameter cannot be used so freely to correlate heat transfer data from systems which differ in configuration but retain a similar basic flow pattern. As an example in Pr ~ 1 fluids basic heat transfer data for flow through circular pipes can be used to predict Nusselt numbers (Nu) for flow parallel to a rod bundle by evaluating the hydraulic diameter of the rod bundle and using this in the non-dimensional correlations for the circular pipe.

This calculation approach is invalid for liquid metal systems, and accordingly theoretical, numerical or experimental heat transfer relationships must be developed to deal with each specific configuration.

The second kind of issues is technological and mostly related to the nature of the HLM-cooled system design and operation. Using coolant chemistry control and surface protective oxide formation to mitigate steel corrosion has consequences in heat transfer performance, particular for the long-term or in abnormal situations, such as build-up of oxides and high level of solid oxide particles. HLM-cooled nuclear reactors usually have open lattice configurations to reduce pumping power needs and enhance passive safety. Flow circulation methods, transients, flow stability and elimination of undesired instability are all important issues to be investigated.

It is necessary to develop and validate more suitable turbulent model(s) for computational thermal hydraulics, especially for complex geometries and critical components, such as the core.

The results of the experiments shall be used, in turn, to improve the related physical models, and to evaluate and benchmark the CFD codes.

5. Compatibility of structural materials with lead

The use of lead or lead-alloy as the coolant in advanced fast reactors implies high-temperature operation and requires structural steels qualified for use in these reactors. Known structural materials like the ferritic/martensitic T91 and the austenitic stainless steel 316L have been a first choice, but they can undergo severe dissolution attack.

Corrosion is, however, but one phenomenon among those relevant to the contact with the liquid metal to be investigated for the qualification of a structural material. Other important phenomena are material failure under static loading, such as brittle fracture, and failure under time-dependent loading, such as fatigue and creep.

5.a Structural materials corrosion in lead

Molten lead or lead-alloy is corrosive towards structural materials. The main parameters impacting the corrosion rate of steels are the nature of the steel (material side), the temperature, the liquid metal velocity and the dissolved oxygen concentration. A provision that can be adopted to reduce leaching out of steel alloying elements (typically nickel which is a component of the austenitic stainless steels and dissolves in the molten lead) is to maintain a controlled amount of oxygen dissolved in the melt. Dissolved oxygen forms a layer of metal oxide on the steel surfaces in contact with lead which protects the steel from dissolution and recovers the metal oxide layer in case of erosion by the flowing heavy metal (self-healing effect).

It has been demonstrated that, in the low temperature range, e.g., below 450 °C, and with an adequate oxygen activity in the liquid metal, ferritic/martensitic and austenitic steels of steels build up an oxide layer which behaves as a corrosion barrier.

However, in the higher temperature range, *i.e.* above ~ 500 °C, corrosion protection through the oxide barrier seems to fail, [5***CHECK REFERENCE***]. Indeed, a mixed corrosion mechanism has been observed, where both metal oxide formation and dissolution of the steel elements occur (Table 5.8).

TABLE 5.8 - Protective action via controlled dissolved oxygen at increasing temperature

<table>
<tr><th>Effective corrosion protection</th><th>Transition zone</th><th>Additional protection needed</th></tr>
<tr><td rowspan="2">Compact stable oxide barrier on ferrite/martensite and austenite</td><td>Oxide formation on ferrite/martensite</td><td rowspan="2">Unstable metal oxide layer

Stable FeAl alloy coating</td></tr>
<tr><td>Mixed corrosion mechanism: oxidation / dissolution on austenite</td></tr>
</table>

400°C 500ºC 550ºC 600ºC

It has been demonstrated that, especially in the high temperature range the corrosion resistance of structural materials can be enhanced by FeAl alloy coating, a recent surface coating technique developed for the purpose and shown effective up to 600 °C.

Several exploratory experiments carried out in the past in LBE and pure lead on different type of steels did show that generally below 450 °C, and with an appropriate dissolved oxygen concentration in the liquid metal, both martensitic and austenitic steels build up an oxide layer (a barrier), which prevents leaching of alloy elements into the liquid metal and liquid metal penetration along grain boundaries. For temperatures above ~ 500 °C, the prevention of liquid metal/steel interaction through the oxide layer seems to fail due to the occurrence of a mixed corrosion mechanisms, where both oxidation as well as dissolution can occur. It has been demonstrated, however, that the corrosion resistance of the structural materials at temperatures above 500 °C can be enhanced by coating the steel surface with FeAl alloys. This is of paramount interest for the fuel cladding, for which coating shall be thin, in order not to significantly affect heat transfer, besides the properties of mechanical stability and adhesion to the substrate material requested to any protective oxide layers. Coating material is FeCrAlY. It contains Al which forms alumina in-situ. To increase the adhesion and improve the stability, such coatings can be melted or fused together with the surface of the substrate, for example by using large area pulsed electron beams as is done in the GESA process, developed by FzK in Germany.

The long–term stability in flowing liquid metal of the oxide layer (for temperatures < 500 °C) as well as the GESA FeAl alloy coating (for temperatures > 500 °C) has not yet been proven, however.

Planned European tests (i.e., the DEMETRA program under IP-EUROTRANS) include therefore corrosion tests in flowing liquid metal (with representative parameters of the fuel cladding and in-vessel components) to estimate corrosion kinetics, and to assess the long–term stability of the protective layers. The experimental program will go along with modeling activities, which help to define the corrosion kinetics for the types of steels under investigation.

It is worthwhile to note that the limitation on the upper temperature of the thermal cycle is considered a temporary compromise solution that allows the reactor design to proceed until new high-temperature materials become available which will allow greater exploitation of the favorable properties of heavy metal coolants; these longer-term developments are likely to hold the key of the commercial viability of advanced fast reactors to be deployed for hydrogen as well as electric energy generation.

Thus the present design approach for ELSY is to limit the mean core outlet temperature to less than 500 °C, and to protect the T91 steel, as the construction material of the unavoidably thermally high loaded fuel cladding tubes, with Fe/Al alloy coating.

5.b Effect of lead on properties of structural materials.

The use of heavy liquid metals, and especially of lead-cooled (Pb) or lead-alloy-cooled (primarily LBE) fast reactor (LFR) concepts of Generation IV requires an assessment of their compatibility with structural materials under the fast neutron spectrum typical of fast reactors. Although western countries did acquire substantial experience with sodium-cooled fast reactors, the expertise on compatibility of stainless steels with sodium is not transferable to lead and lead alloys, owing to the significant differences in their physical and metallurgical properties. Thus, the older literature dedicated to the mechanical properties of steels, from carbon steels to high Cr steels, in contact with lead and lead alloys is essentially of Russian origin. The Russian research on HLM technology will soon cover one century, first oriented toward developing fundamental understanding of the liquid metal embrittlement (LME) mechanism, then largely at the beginning of the fifties, with the development of submarine propulsion reactors in parallel with two full scale ground test reactor facilities, using LBE as a coolant.

Because the 316L austenitic stainless steel and the T91 martensitic steel have been pre-selected for the design of future European transmutation facilities (EFIT, XT-ADS) and eventually also for the lead-cooled ELSY, the effect of LBE or lead on the mechanical behavior of these steels is being extensively investigated in

Europe and worldwide, and results are available, but they are not yet exhaustive. The effect of lead or LBE on the tensile properties of T91 is well documented, thanks to the European TECLA and MEGAPIE-TEST programmes of FP5. It has been shown that under MEGAPIE relevant conditions, (*i.e.,* temperatures below 400 °C and very low dissolved oxygen concentration in the liquid metal, *i.e.,* reducing conditions below the potential of iron oxide formation), not only no oxide barrier forms on the steel (*i.e.,* direct contact between steel surface and liquid metal), but worsening of the mechanical performance of the steel also occurs, if surface cracks are present. Particularly, Low Cycle Fatigue (LCF) tests performed on T91 samples with different pre-treatments, did show that strong LBE attack appreciably reduces the LCF resistance of the steel, with respect to pre-oxidized samples and to the as-received samples. These tests as well as slow strain rate and tensile tests provide evidence of the effect of the surface condition on the mechanical behavior of the T91 when in contact with LBE. The latest results of LCF tests performed on the AISI 316L steel in air and LBE have shown a fatigue lifetime reduction in LBE for higher strain ranges.

Today, and in spite of a lack of quantitative results on fatigue and fracture, based on analysis of the data collected on the tensile and fatigue tests, the question of the susceptibility to LME (embrittlement) of T91 in contact with LBE can be addressed, particularly how to proceed from the metallurgical and chemical points of view to prevent LME. Thus, there are some data available on real systems about wetting, which is one of the two main conditions for occurrence of LME, in addition to abundant theoretical literature. The knowledge of the stress level responsible for plastic deformation, even at microscopic scale, as the second main condition for occurrence of LME, would allow eventually the definition of the criteria for preventing LME failure. Environment-assisted cracking (EAC) is a phenomenon closely related to LME, which permits an interpretation of the results of some tensile tests conducted on T91 or 316L steel in lead or LBE environments.

Information on the effect of lead or LBE on the creep properties of both T91 and 316L is scarse. There is almost no information on creep strength, creep damage and creep crack growth rates in the currently available accessible literature.

Proposed explanations consider liquid metal accelerated creep (LMAC), whereby liquid metals can accelerate, at the same time, creep and the nucleation growth of vacancy voids near the metal surface in traction or compression.

Creep tests were performed in Russia in the context of the BREST-0D-300 reactor system, on a chromium steel 10Kh9NSMFB (containing 1.2%Si) in flowing liquid lead under 70 to 100 MPa between 420 and 550 °C, showing an earlier transition into the third stage of creep and a decrease of the duration of the steady creep stage, explained as a consequence of the lead corrosiveness. More detailed information about the test conditions and composition and structure of the steel/lead interface would be of interest at all stages of the test. This would imply access to the Russian literature on structural materials in contact with liquid metals for nuclear applications.

Information on fracture mechanics, from fracture toughness to crack growth behavior in contact with LBE, does not exist. There is a large body of literature devoted to the fracture of structural materials, hardened and often embrittled under irradiation, covering a wide range of experimental test conditions. This is, indeed, information of primary importance, which, for example, did allow for an estimate of the service life of the MEGAPIE target window. There is no such information available for structural materials in contact with lead. It is sometimes stated that the ductile to brittle transition temperature, which may increase by approximately one hundred degrees after irradiation, should be only little increased by contact with HLMs. If proven, this fact would be of paramount importance and hence it must be verified experimentally.

In summary, dedicated test plans will have to be set up in order to provide data, particularly in the higher temperature range, on tensile, creep, creep-fatigue and fracture mechanics and fatigue crack growth of steels in contact with the selected HLM, including testing of irradiated specimens. Results of similar experiments carried out in the frame of different technologies, such nuclear fusion, shall not be disregarded as they can be conveniently used as a guide for the HLM-dedicated experiments. Austenitic stainless steels (T ≤ 500 °C) and ferritic/martensitic steels (T ≤ °600 C) are likely to remain the main candidate materials for the future power reactors in spite of uncertainties in the areas of irradiation induced embrittlement at low temperatures and radiation damage from high He/dap ratio. Prospective candidate materials are the ODS Martensitic steels (temperature window can be increased to 650 °C), and in the longer term the ferritic ODS steels, ceramic composites and refractory alloys for the higher temperatures.

6. Core

The LFR core design approach is here presented and discussed. An integrated neutronic-thermal/hydraulic approach is envisaged in order to address the whole process towards the most effective solution to what concerns the design goals, according to the related technological constraints.

Starting from the peculiarities of lead- or LBE-cooled systems, the overall design approach will be presented and actualized according to general typologies of reactors: critical, subcritical (*i.e.,* ADS) and "adiabatic". The latter typology stands for critical reactors able to operate while maintaining unaltered the inventory of "valuable" isotopes (e.g.: all TransUranics, TRUs) along the cycle (in an "extended" nuclear equilibrium state), thus not exchanging with the environment (hence the term adiabatic) any bulk materials except either Natural or Depleted Uranium as an input stream, and Fission Products (FPs) as output. This solution, implementing the fuel cycle closure within the reactor itself, represents the most charming candidate for giving body to the sustainable nuclear development aimed by the Generation-IV initiative.

6.a Introductory remarks for LFR core design

Besides the main changes in the neutronic performance of the system, due to the hard spectrum set up by lead (see section 5.3), several other peculiar aspects must be accounted for in approaching the core design of a LFR.

These peculiarities, mainly related to the thermal/hydraulics properties of the coolant and the specific technological constraints introduced by the choice of lead, have to be added to the list of criteria and specimens commonly considered in the standard design route of FRs, for a proper core conceptualization of such systems.

6.a (i) Preliminary evaluation of lead and LBE impact on core design

The low[6] heat removal capability of lead (about one tenth with respect to sodium, see section 5.4) imposes the need of large coolant channels to ensure the proper cooling of the system, immediately leading to low power densities in the core due to the high coolant volumetric fraction in the elementary cell (as mentioned in section 5.1.a). The first drawback impacts the overall dimensions of the core, which – on the other hand – has to comply with the need to reduce the system volume because of seismic risk. The second drawback is related to the specular reduction of the fuel volumetric fraction in the cell, which results in the need of increasing the cell reactivity (either by increasing the number of pins – modifying as a consequence also the system power – or the fuel enrichment – hence affecting the breeding capabilities). It must be noticed that the absolute magnitude of the latter effect is limited by the low absorption cross sections of lead (see section 5.3.a).

6.a (ii) Technological constraints for LFR design

The first constraint following the choice of lead as coolant is related to the high melting point (327 °C) of the latter, resulting in a lower limit on coolant inlet temperature T_{inlet}. The lower limit on coolant inlet temperature is moved to 123.5 °C in case of LBE since its lower melting point.

On the other hand, the high corrosion of Ferritic Martensitic Steels (FMS, preferable to austenitic Stainless Steels, SS, for neutronics since the lower Ni content) in the lead environment (see section 5.5a) imposes an upper limit on coolant outlet temperature T_{outlet} and/or on the ratio between the maximum linear power q' and the fuel pin diameter d (determining the thermal head between the clad and the coolant to evacuate the fission power from the fuel pin). Recalling section 5.5.a, the maximum wall temperature for the clad T_{clad} must be kept within 500 °C to prevent corrosion unless superficial coatings are foreseen: preliminary evaluations

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[6] Here “low” is intended with respect to other traditional heat removal vectors such as water and sodium.

on the GESA aluminization technique seem to confirm good resistances to corrosion up to 600 °C under active oxygen control.

Structural integrity must be preserved also against erosion phenomena: an upper limit of 2 m/s on the coolant flow velocity v through the channel must be therefore taken into account. As for corrosion, surface coatings may allow for maximum coolant velocities up to 3 m/s.

The effect of neutron irradiation on FMS imposes a further lower limit on the coolant inlet temperature to mitigate embrittlement of the structures (section 5.5.b). A typical value for the minimum allowed temperature is set to 400 °C.

Finally, the hard spectrum of LFRs imposes stringent constraints to what concerns the structural damage (the number of Displacements per Atoms, DpA, indeed depends mainly on the hard tail of the neutrons spectrum). According to this, the in-pile residence time for Fuel Assemblies (FAs) made of FMS must be set to keep the fast neutron (E > 0.1 MeV) fluence below $4 \cdot 10^{23}$ n/cm: hence the corresponding limit on maximum irradiation in a LFR might force a reduction in the planned fuel burnup performance.

6.b Conceptual design approach

Core design aims at determining the main parameters which univocally define a reactor configuration providing the required neutronic features and complying with all the (mainly) thermal/hydraulic constraints (among these, the ones specific to lead are listed in section 5.6.a (ii)). Since there is a strong inter-dependence among the core parameters, it is therefore a complex task to balance the pros and cons considering any consistent combination of these parameters.

If one defines a "reactors space" as a hyper-space where the axes represent the independent core parameters, core design can be visualized as the research for an optimal operating point in this multi-parameter diagram: the technological limits introduce boundaries narrowing the viability domain, the parameter interdependence laws define hyper-surfaces representing the relationships between degrees of freedom and constraints, and the goal features provide the criteria to orient the choice for the most suitable operating point in the design domain.

6.b (i) Critical reactors

In order to define the viability domain in the reactor space, each single constraint must be translated into an equivalent inequality defining the actual range for the corresponding parameter. In particular, the ranges for the coolant inlet temperature and the clad wall temperature are fixed according to

$$T_{inlet} = \max\{T_{melt}, T_{embrittlement}\} + \Delta T_{margin} \qquad (5.12)$$

$$T_{clad} = T_{corrosion} - \Delta T_{margin} \tag{5.13}$$

The coolant flow velocity must be analogously limited according to

$$v \le v_{erosion} \tag{5.14}$$

Besides the specific constraints of LFRs, the maximum fuel temperature must be accounted for to prevent (with a sufficient margin) the possibility of fuel melting. According to this, the maximum fuel temperature can be translated into an equivalent upper limit on the maximum linear power, expressed by the conductivity integral relation:

$$q'_{\max} \le 4\pi \int_{T_{f-g}}^{T_0} k_f(T)\,\mathrm{d}T \tag{5.15}$$

where $T_0 = T_{f,melt} - \Delta T_{margin}$ is the maximum allowable temperature for the fuel (at the center of the pellet), $T_{f\text{-}g}$ is the temperature at the pellet surface (*i.e.,* at the fuel-gap interface) and $k_f(T)$ is the thermal conductivity of the fuel.

Once the viability domain in the reactor space has been determined, the starting point for the neutronic design of a core is represented by the thermal/hydraulics design of the fuel pin and the coolant channel: as a matter of fact, both the pin radius r and lattice pitch p depend only on the thermal-hydraulic consistency of the system. Since the system must be dimensioned to prevent out-of-range working everywhere in the core, the most peaked fuel pin is assumed as reference for the design.

The core inlet and outlet average temperatures of the coolant can be identified according to the technological constraints introduced in section 5.6.a (ii), so that the maximum outlet temperature in the hottest channel can be inferred in turn by introducing the expected estimate for the radial distribution factor f_{rad}[7].

A preliminary evaluation of the active height h_{fuel} can then be introduced together with the gas plenum height h_{plenum} and, according to the latter, the clad thickness s_c determined to stand the pressure of gaseous FPs corresponding to the aimed BU.

By introducing an attempt value for the axial maximum-to-average factor f_{ax} (guessed or borrowed from previous calculations or analogous systems), an estimate of the power distribution along the pin can be inferred by assuming

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[7] The “radial distribution factor” can be defined as the ratio between the power released from the hottest pin to the average pin power in the core. It should be noticed that uneven outlet temperature distributions are usually mitigated (by gagging the inlet orifice of wrapped FAs and/or by segmenting the core into zones with differently enriched fuels or different fuel volumetric fractions to flatten the power distribution) not to damage the thermal efficiency of the system.

$$q'_{\max}(z) = q'_{\max}\int_0^{h_{fuel}} \cos\left(\frac{\pi}{L}\left(z - \frac{h_{fuel}}{2}\right)\right)\mathrm{d}z \tag{5.16}$$

The parameter L in eq. (5.16) must be computed so that the length of the arc over which the cosine is defined provides the assumed axial distribution factor, that is:

$$\int_{-\beta}^{\beta} \cos(\alpha)\mathrm{d}\alpha = \frac{1}{f_{ax}} \rightarrow \beta = \arcsin\left(\frac{1}{2f_{ax}}\right) \Rightarrow L = \frac{\pi h}{2\beta} .$$

The coolant temperature profile along the hottest channel, $T_l(z)$, can then be retrieved as

$$T_l(z) = T_{inlet} + (T_{outlet} - T_{inlet})f_{rad}\frac{\int_0^z q'_{\max}(z')\mathrm{d}z'}{\int_0^{h_{fuel}} q'_{\max}(z')\mathrm{d}z'} \tag{5.17}$$

At last, the gap can be preliminarily dimensioned, s_g, to host the fuel swelling related to the aimed BU, not to incur Pellet-Clad Mechanical Interaction (PCMI).

In order to guarantee the respect of the clad wall and fuel centreline temperature limits, the dimensioning of the fuel pin (or, equivalently, of the fuel pellet since all remaining radial dimensions have been already fixed), together with the evaluation of the maximum admissible linear power, can then be carried out by taking into account the thermal fluxes through the pin. As a matter of fact, the thermal resistance of the fuel, the gap and the clad act in series in determining the succession of temperature gains providing the necessary thermal heads to evacuate the local fission power from the fuel to the coolant, in the typical temperature profile of Figure 5.13.

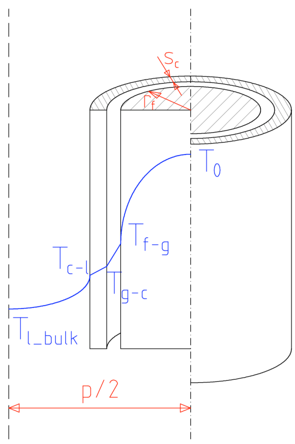


Fig. 5.13 - Radial temperatures profile in the elementary cell.

All these gains depend on the geometry and the materials of the pin only, as expressed by

$$T_0(z) - T_{f-g}(z) = \frac{q'_{\max}(z)}{4\pi\langle k_f \rangle} \tag{5.18}$$

$$T_{f-g}(z) - T_{g-c}(z) = \frac{q'_{\max}(z)}{2\pi k_g} \ln\frac{r_g}{r_f} \tag{5.19}$$

$$T_{g-c}(z) - T_{c-l}(z) = \frac{q'_{\max}(z)}{2\pi k_c} \ln\frac{r_c}{r_g} \tag{5.20}$$

$$T_0(z) - T_{f-g}(z) = \frac{q'_{\max}(z)}{2\pi\, h_l\, r_c} \tag{5.21}$$

In the previous system, single subscripts refer to materials (*f* for fuel, *g* for the gas filling the gap, *c* for clad and *l* for the coolant) while coupled subscripts refer to materials interfaces (*f-g* for fuel-gap interface, *g-c* for gap-clad interface and *c-l* for clad-coolant interface); k_i indicates the thermal conductivity of material *i* and h_l indicates the heat-transfer coefficient of the coolant.

It is therefore possible to put together eqs. (5.18-5.21) in order to obtain a single expression for the dimensioning of the pin (which can be limited to r_f):

$$\begin{aligned} T_0(z) = T_l(z) + \frac{q'_{\max}(z)}{2\pi}\Bigg(& \frac{1}{2\langle k_f \rangle} + \frac{1}{h_g r_f} + \\ & + \frac{1}{k_c}\ln\frac{r_f + s_g + s_c}{r_f + s_g} + \frac{1}{h_l (r_f + s_g + s_c)}\Bigg) \end{aligned} \tag{5.22a}$$

The same relation holds for hollowed fuel pins too, with minor changes: said γ_f the ratio between the hollow and pellet radii, equation (5.22a) becomes

$$\begin{aligned} T_0(z) = T_l(z) + \frac{q'_{\max}(z)}{2\pi}(1-\gamma_f^2)\Bigg(& \frac{1}{2\langle k_f \rangle}\left(1 + \frac{\gamma_f^2 \ln\gamma_f^2}{1-\gamma_f^2}\right) + \\ & + \frac{1}{h_g r_f} + \frac{1}{k_c}\ln\frac{r_f + s_g + s_c}{r_f + s_g} + \frac{1}{h_l (r_f + s_g + s_c)}\Bigg) \end{aligned} \tag{5.22b}$$

The logical process for the fuel pin dimensioning can be represented by the dependencies scheme of Figure 5.14.

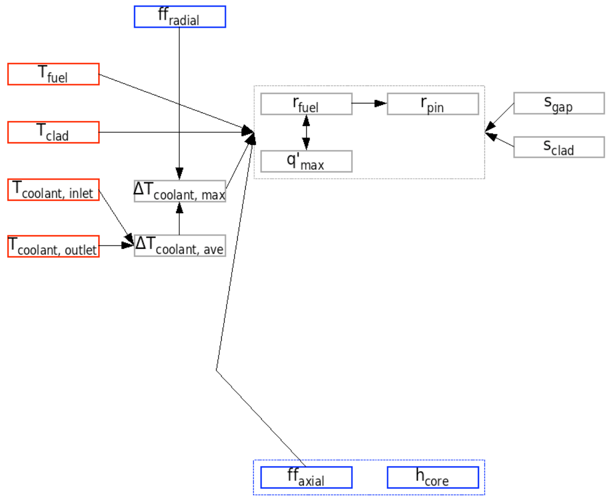


Fig. 5.14 - Scheme of core parameters dependences for fuel pin diameter dimensioning.

Once the fuel radius (and the eventual hollow one) has been defined, the coolant channel must be dimensioned. The pitch of the pins lattice, p, is to be chosen according to the average coolant temperature gain along the average channel. The increase of the coolant temperature in the channel is derived from the enthalpy balance equation

$$\rho_l A v c_p \Delta T = \frac{q'_{\max}}{f_{rad} f_{ax}} h_{fuel} \tag{5.23}$$

where ρ_l and c_p are the coolant density and heat capacity respectively (see section 5.4.a) and A, the flow area of the channel, depends on the fuel pin radius and lattice pitch according to

$$A = \begin{cases} \dfrac{p^2\sqrt{3}}{2} - \pi r_c^2 & \text{hexagonal lattice} \\ p^2 - \pi r_c^2 & \text{square lattice} \end{cases} \tag{5.24}$$

The reference pitch value must be set by taking into account also the maximum allowed coolant velocity (5.14) and the pressure drops through the channel.

The latter comes from the requirement of providing a sufficient natural circulation, in case of Unprotected Loss Of Flow (ULOF) accident, so as to guarantee nominal heat removal from the core within an acceptable temperature range

ΔT_{ULOF}[8]. According to this, the channel must be dimensioned so as to keep the pressure losses in.

The thermal head assessing in natural circulation can be easily determined as

$$\Delta p = \Delta\rho g h_{buoyancy} \cong 3\alpha \Delta T \rho g h_{buoyancy} \tag{5.24}$$

where g is the strength of the gravitational field, $h_{buoyancy}$ is the buoyancy height (*i.e.* the height of the primary circuit hot-leg, from core midplane to steam generators midplane) and α is the linear thermal expansion coefficient of the coolant.

This forcing term must overtake the pressure drop through the whole primary circuit, expressed, separating the contribution within and outside the core, as

$$\Delta p = \Delta p_{core} + \Delta p_{system} = f \frac{h_{channel}}{D_h} \frac{\rho v^2}{2} + \Delta p_{system} \tag{5.25}$$

where f is the effective friction term in the channel, D_h and $h_{channel}$ are, respectively, the hydraulic diameter and the length of the channel.

Putting together eqs. (5.24) and (5.25), and applying eq. (5.23) to ULOF case, the following expression involving the geometry of the channel can be extracted

$$3\alpha \Delta T \rho g h_{buoyancy} = f \frac{h_{channel}}{D_h (1 + 3\alpha \Delta T)} \left(\frac{q}{A c_p \Delta T} \right)^2 + \Delta p_{system} \tag{5.26}$$

for testing whether the temperature gain set up for coolant circulation is acceptable.

The logical process of coolant channel dimensioning can be represented by the dependencies scheme of Figure 5.15.

[8] ΔT_{ULOF} is the temperature gain along the channel which establishes in order to provide the required prevalence for natural circulation. According to core integrity, it is important that ΔT_{ULOF} settles so as to keep T_c below the critical clad melting temperature: as a matter of fact, the allowed temperature gain in accidental condition is higher than the normal one because the higher temperature range is supposed to last for a limited time span (typically 30 minutes before human intervention), during which erosion/corrosion constraints can be neglected.

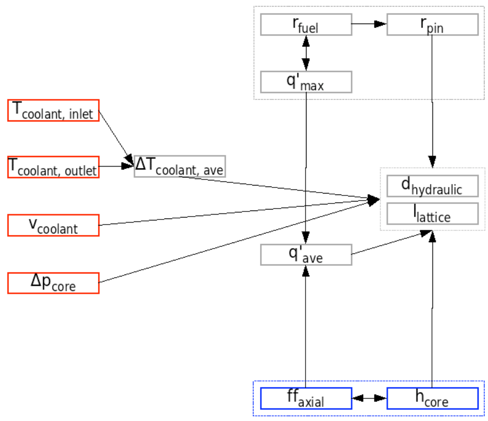


Fig. 5.15 - Scheme of core parameters dependences for coolant channel dimensioning.

Once the elementary cell has been determined, the axial and radial form factors are used to infer the average linear power in the core:

$$\langle q' \rangle = \frac{q'_{\max}}{f_{rad} f_{ax}} \tag{5.27}$$

This can be used in turn to calculate the total development of the fuel H needed to achieve the desired nominal power P_{th}:

$$H = \frac{P_{th}}{\langle q' \rangle} \tag{5.28}$$

Combining the total development of the fuel H with the preliminary core height h_{fuel} to retrieve the number of fuel pins n_{pins}, the radius of the core equivalent cylinder results

$$r_{core} = \sqrt{\frac{1}{\pi} n_{pins} A} = \begin{cases} \sqrt{\dfrac{H}{n_{pins} h_{fuel}} \dfrac{p^2 \sqrt{3}}{2}} & \text{hexagonal lattice} \\ \sqrt{\dfrac{H}{n_{pins} h_{fuel}} p^2} & \text{square lattice} \end{cases} \tag{5.29}$$

The aimed BU performances allow also to preliminary evaluate an in pile residence time for the fuel. According to this, the core can be also segmented into batches for refueling, so to define the mean fuel ageing at Beginning of Cycle

(BoC) and End of Cycle (EoC), averaging the in pile residence time of the FAs belonging to different batches just before (EoC) and immediately after (BoC) the refueling. This approach leads to a 1-batch strategy approximation, which has been proven [Ref.X][NEED TO INCLUDE REFERENCE] to be equivalent – in terms of criticality swing along the cycle – for the criticality analysis of the core.

The assessment of criticality can be performed taking into account that the overall shape of the system fixes the geometrical buckling of the reactor. Considering the system as a homogeneous volume V, the neutrons net balance, expressed as the ratio of the material buckling upon the geometrical one, can be translated into a balance between the net production in the reactor over the net leakage from the latter:

$$\frac{Prod}{Leak}=1 \quad \Rightarrow \quad \frac{\int_V \nu\Sigma_f \phi \,\mathrm{d}V - \int_V \Sigma_a \phi \,\mathrm{d}V}{\int_V \nabla\phi \,\mathrm{d}V}=1 \tag{5.30}$$

The volumes in the cells are fixed by the thermal/hydraulic analysis of the channel. The neutron spectrum is therefore also fixed by the volumetric fraction of coolant, fuel and structural materials in the cell. For criticality, neutronic calculations must be performed to assess the composition of the fuel (*i.e.,* its enrichment), which is used as an almost free parameter to match the required reactivity during the cycle and the power distribution flattening: as a matter of fact, the fuel must be enriched so to adjust the material buckling coherently with the geometrical one. The increase of the fuel enrichment both acts in increasing the fission term and in reducing the absorption one (*i.e.,* the fissile is added to the detriment of the absorbing fissile).

It is clear that modifying the mutual abundances of fissile and fertile also changes the breeding capacity of the system, which could represent a design goal acting as feedback parameter in the design process; the same is valid for the eventual dispersion of MAs in the fuel. Furthermore, it is to be noticed that the enrichment also determines the flux level, according to the fixed power density in the fuel: the higher the enrichment, the lower the flux needed to achieve the same power density

$$q' = \pi r_f^2 \phi Q \rho_f \sigma_f \tag{5.31}$$

For instance, in designing experimental reactors, the flux level could represent a binding criterion: again, also the peak neutron flux can be used as a feedback parameter for core design.

As a matter of fact, the collection of output performances resulting by the present core configuration should be used as feedback to adjust the core design in order to achieve completely all the aimed goals, in an iterative process.

The overall dependencies scheme for core design is shown in Figure 5.16.

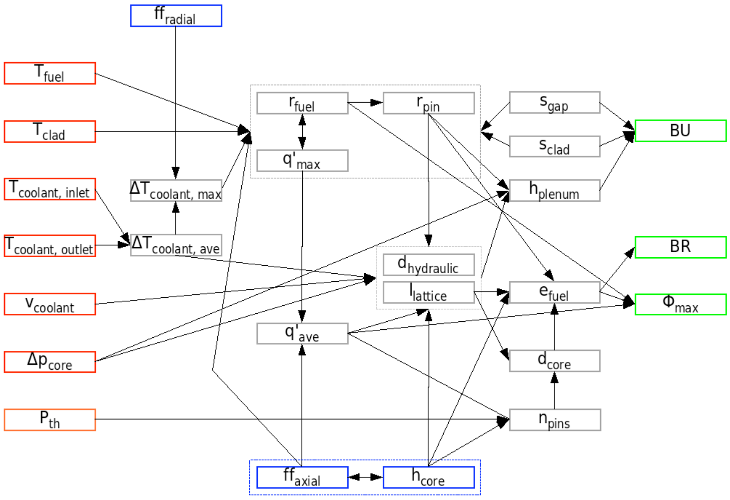


Fig. 5.16 - General scheme of core parameters dependences.

Case study: ELSY

The technological constraints in ELSY design were fixed as:

- max$\{T_0\}$ = 2100 °C to prevent traditional MOX fuel melting;
- T_{inlet} = 400 °C for limiting structurals embrittlement;
- T_c = 550 °C for limiting corrosion under active oxygen control;
- v = 2 m/s for limiting erosion.

An acceptable value of the coolant outlet temperature was also set to 480 °C.

Under such hypotheses, and assuming also that in case of unprotected transient due to Unprotected Loss Of Flow accident (ULOF) the cladding is allowed to reach a maximum temperature of 700 °C when assessing natural circulation, the iterative design process led to the assessment of the core parameters. The resulting parameters are listed in the table below.

TABLE 5.8 - First issue of ELSY core parameters

Parameter	Reference value
Fuel pellet (solid) radius	4.50 mm
Gap thickness	0.15 mm
Clad thickness	0.60 mm
Fuel pin radius	5.25 mm
Pins lattice pitch (square)	13.9 mm
Active height	90 cm
Coolant velocity	1.61 m/s
Maximum linear power	347 W/cm

The overall core layout, developed in ENEA, resulted by arranging all the fuel pins needed to achieve the aimed thermal power (1500 MWth) according to square 21x21 pins patterns. The resulting 162 Fuel Assemblies (FAs) have been organized to reproduce the pseudo cylindrical core shown in Figure 5.17.

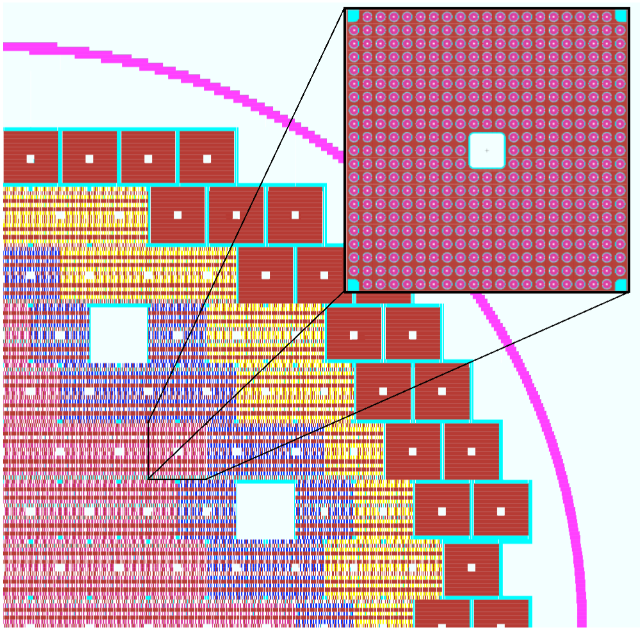


Fig. 5.17 - Preliminary ELSY core scheme (1/4) surrounded by dummy elements within the barrel and FA layout.

The enrichments needed to ensure both criticality and power/FA distribution flattening (with a 1.2 limit for maximum-to-average power/FA ration) for the reference core led also a unitary Breeding Ratio (BR), which permits a moderate criticality swing during the cycle, limiting therefore the anti-reactivity required to compensate it. On the other hand, the smear density of the fuel within the clad (0.89) allows a maximum fuel BU of 60 GWd/tHM, which is below the aimed value (100 GWd/tHM).

Considering the BU goal priority with respect to the BR one, the smear density had to be reduced to 0.84 in order to obtain the aimed BU. The adopted solution was to hollow the fuel pellet (2 mm hole diameter) without altering any other core parameter.

The removal of part of the fuel resulted in an increase of the enrichments to maintain the criticality. In first approximation (*i.e.* neglecting the reactivity gain due to the fewer captures by the lower U amount), it could be thought to create the hole in the pellet by removing selectively only U, leaving unaltered the Pu amount by increasing its content (i.e., enrichment) in the remaining fuel. Within the clad, the total fuel volume is thus reduced by a factor

$$\frac{V'_f}{V_f} = \frac{\pi\left(r_f^2 - r_h^2\right)}{\pi r_f^2} = \frac{4.5^2 - 1.0^2}{4.5^2} = 0.95 \tag{5.32}$$

which in turn can be ascribed to an equivalent loss of U only, according to

$$\frac{V'_U}{V_U} = 1 - \frac{1 - \frac{V'_f}{V_f}}{VF_U} = 1 - \frac{0.05}{0.83} = 0.94 \tag{5.33}$$

The achievement of the target BU implies thus a degradation of the BR according to the same relative reduction of U amount: as expected, the BR for the new system was found to be 0.94.

Case study: ENHS

In the ENHS design scheme all the main dependences among the core parameters have been referred to the core pitch-to-diameter (*p/d*) ratio, inheriting the same approach of thermal reactors design. The desirable BR is therefore achieved by adjusting the core *p/d* ratio; moreover, the larger is the *p/d* ratio, the smaller becomes the coolant friction losses through the core and the larger becomes the power that can be removed from the core by natural circulation.

The technological constraints in ENHS design were fixed as:

- $\max\{T_0\}$ = 900 °C to prevent metallic fuel melting;
- T_{inlet} = 400 °C for limiting structurals embrittlement;
- T_c = 600 °C for limiting corrosion.

An acceptable value of the coolant outlet temperature was also set to 550 °C.

The natural circulation goal can be achieved, smoothing the constraints to the core design, by tuning the height of the riser in order to adjust the thermal head to the actual pressure drops.

The core design process led to the following core parameters.

TABLE 5.9 - First issue of ENHS core parameters

Parameter	Reference value
Fuel pellet (solid) radius	5.630 mm
Gap thickness	0.870 mm
Clad thickness	1.300 mm
Fuel pin radius	7.800 mm
Pins lattice pitch (hexagonal)	21.216 mm
Active height	125 cm
Riser height	13 m
Coolant velocity	0.44 m/s
Maximum linear power	179 W/cm

The overall core layout resulted by arranging all the fuel pins needed to achieve the aimed thermal power (125 MWth) according to a uniform, core-wise hexagonal lattice. No FA has been introduced in the model, because of the modularity of the core: fuel pins are therefore directly connected to the lower diagrid to reproduce the aimed pseudo-cylindrical core configuration. Figure 5.18 shows the ENHS battery module layout where both the core and the coolant riser are represented as resulting from the core design.

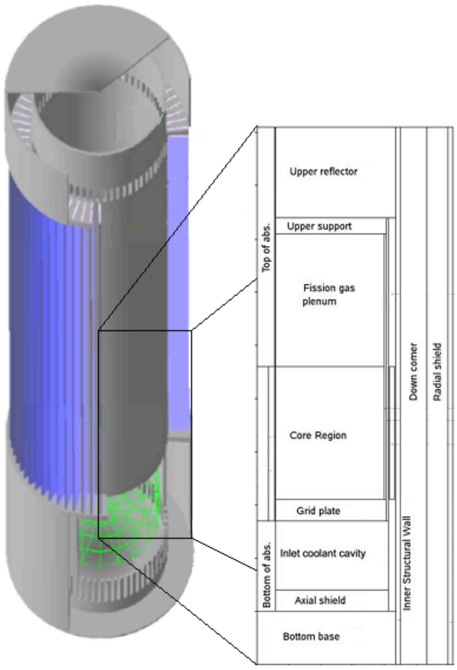


Fig. 5.18 - ENHS battery module layout and simplified sketch of its central region.

The enrichments needed to ensure the criticality for the reference core led to an insufficient BR (0.97) with respect to the aim of $\Delta k_{\mathrm{eff}} \sim 0$. In order to increase the BR, the *p*/*d* parameter had to be adjusted to permit a reduction of the Pu amount, thus a higher U amount and, in turn, a higher BR.

The reduction of the fissile inventory can be obtained by increasing the intrinsic reactivity of the system, *i.e.* a reduction of the capture losses in the coolant.

The aimed BR (1.02) was then obtained by moving to a lower *p*/*d* ratio (reducing the pitch): the optimal *p*/*d* ratio resulted then 1.34.

The smaller flow area in the fuel cell (thus the higher pressure drops through the core) set the coolant velocity to a lower value, necessary to increase the driving ΔT (clad wall - coolant bulk) and thus the prevalence due to the thermal head.

Nevertheless, the small reduction of the pitch required to adjust the BR (from 21.216 to 20.904 mm) allows a clad temperature still below the safety limits.

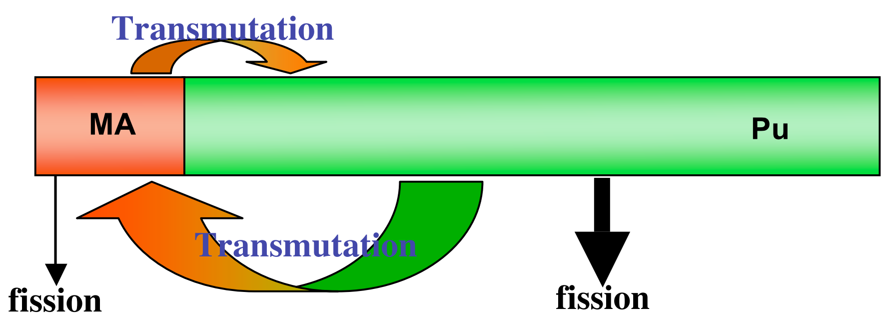
6.b (ii) Subcritical reactors

The recent interest in Accelerator Driven Subcritical (ADS) systems has been driven by the possibility of setting up power reactors able to burn considerable quantities of MAs by eliminating the safety related drawbacks due to the inclusion of MAs in the fuel: as a matter of fact, relying on a large margin to criticality allows to neglect the reduction of the delayed neutrons fraction of such systems, leaving room to increase the MAs content – from a neutronic point of view – at will.

The design of an ADS core must therefore take into account the two aims at optimizing the MAs burning capabilities and producing energy: about these two main points the logics of the core design must be exploited to highlight the rationales for answering three basilar questions:

1. what exactly means burn MAs “at best”?
2. how the burning capability can be optimized?
3. what about the two main goals, should they be contradictory?

A first consideration can be brought taking into account that the fuel of an ADS will be composed of a “driver” material, containing Pu, which is the main responsible for the criticality of the system, and the “target” MAs. The mutual abundances of Pu and MAs in the fuel set both fission and transmutation reaction rates: according to Figure 5.19, the higher the MAs content, the higher the reaction rates of MAs to Pu transmutation (overcoming the opposite Pu to MAs transmutations).



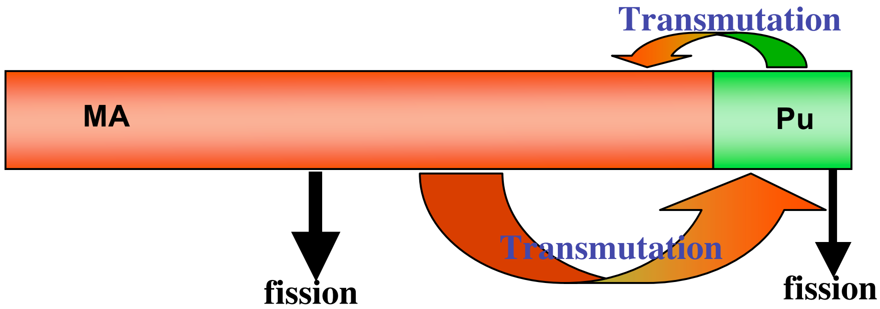


Fig. 5.19 - Fission and transmutation rates as a function of Pu and MA mutual abundances in the fuel.

Another important remark must be made to what concerns the normalization of the reaction rates in the system: according to the fission Q-value, for every TWh of energy produced, some 42 kg of fuel is actually fissioned, the contributions of Pu and MAs being split according to their mutual abundances. On the other hand, a different MA disappearance rate can be observed, representing the further net contribution of MAs to Pu transmutation: hence MA removal rates higher than 42 kg/TWh must be interpreted as 42 kg MAs actually fissioned, and the remaining transmuted into new Pu (the system would therefore act as a Pu breeder); *vice-versa*, MAs removal rates lower than 42 kg/TWh imply that the complement is represented by Pu fissions, reducing the inventory of the latter (the system would also act as a Pu burner).

It is worth noticing that the rate of Pu production/removal directly impacts also the criticality swing along the cycle: according to this, and taking into account the accumulation of poisoning FPs during operation, the goal of zeroing the criticality swing points at a Pu/MAs assortment in the fuel different from the one needed to obtain an equilibrium Pu content.

The last preliminary consideration takes into account that the Pu content in the fuel is set according to the required reactivity inventory for the system: assuming that the pin lattice is almost fixed because of thermal/hydraulic design (as for critical reactors, see previous subsection), the power size of the reactor can be translated into its geometrical size. According to this, the more pins are arranged in the core, the lower the Pu content in the fuel to provide the aimed reactivity, as shown in Figure 5.20.

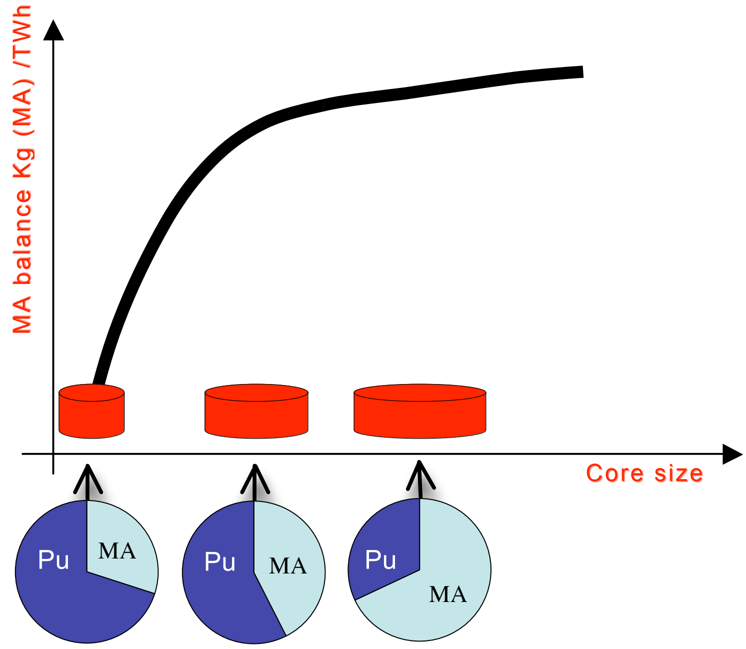


Fig. 5.20 - Fuel enrichment as a function of core size.

Figure 5.20 also highlights a direct relationship between the core power and the MAs burning performances: bigger cores allow the arrangement of a higher relative amount of MAs in the fuel, thus provide higher MAs removal rates.

According to these introductory remarks, the previous three main questions can be answered, providing the final design strategy for an ADS core:

- unless precise policies for producing new Pu are envisaged, the burning of MAs at best means that no "expensive" neutrons must be used to either burn or breed Pu, thus devoting all the net fission losses to MAs (-42 kg/TWh of MAs and 0 kg/TWh Pu balances, according to the s.c. "42-0" approach);
- since looking for a MAs burning performance better than -42 kg/TWh is meaningless, the optimization leads to the research of the minimum cost of the TWh or, considering the velocity of burning 42 kg/h per TW, the minimum cost of the deployed power (which is the same optimization required for the energy production).

Exploiting these general remarks, the design approach for an ADS core immediately follows, as depicted in Figure 5.21.

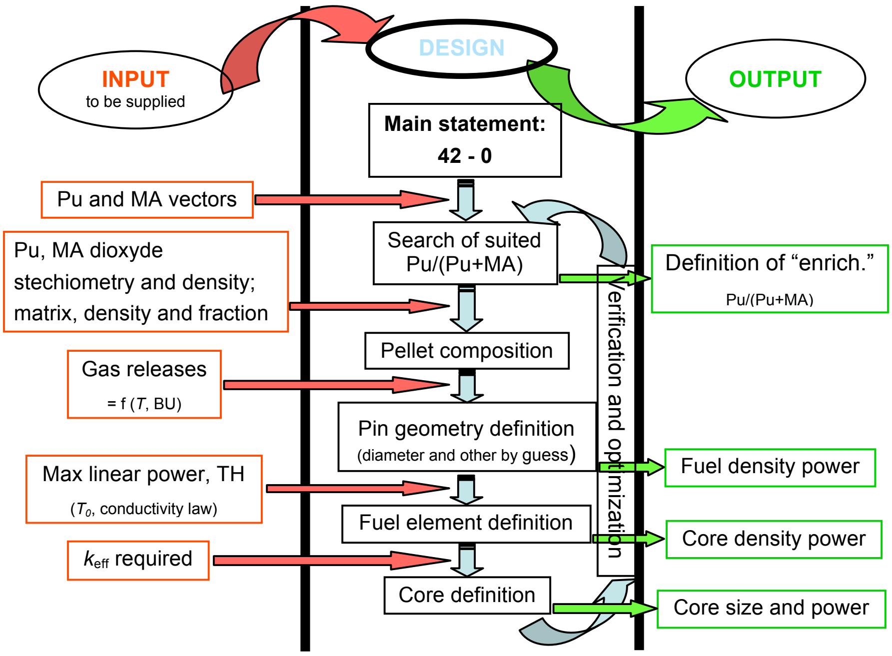


Fig. 5.21 - ADS core design scheme.

The mutual influence between the main core parameters (roughly shown in Figure 5.22, where the arrows thickness represent the strength of the corresponding interdependences) can be better visualized into a properly compiled worksheet ("A-BAQUS"), organized according to the general logic scheme of Figure 5.21.

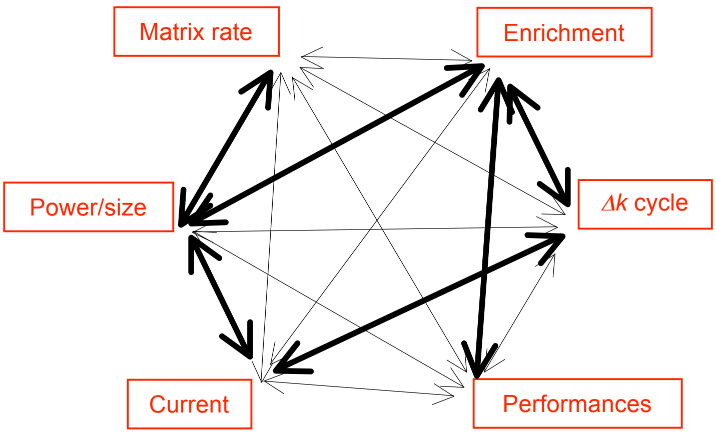


Fig. 5.22 - Mutual interdependences between main core parameters.

The A-BAQUS worksheet is organized per quadrants, with multiple axes representing the main core parameters. In the first quadrant the two axes represent the relative Pu content in the fuel (thus the fuel enrichment *e*) and the percentage of inert matrix in the pellet, as shown in Figure 5.23.

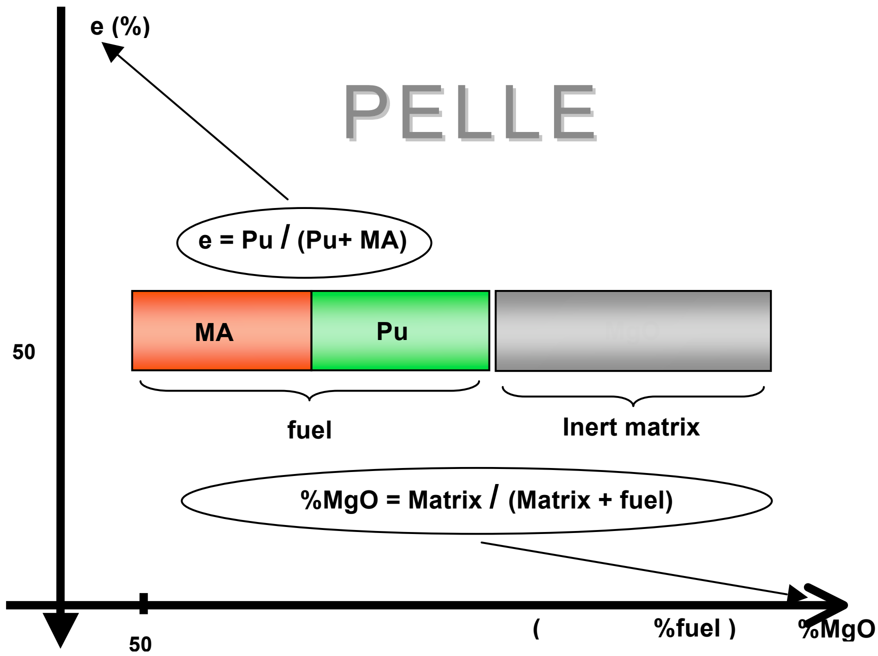


Fig. 5.23 - First quadrant of the A-BAQUS worksheet.

In the same quadrant it is possible to add two more axes, relating the fuel enrichment to the MAs transmutation capabilities (Figure 5.24) and the criticality swing along the cycle (Figure 5.25) as described in the previous lines.

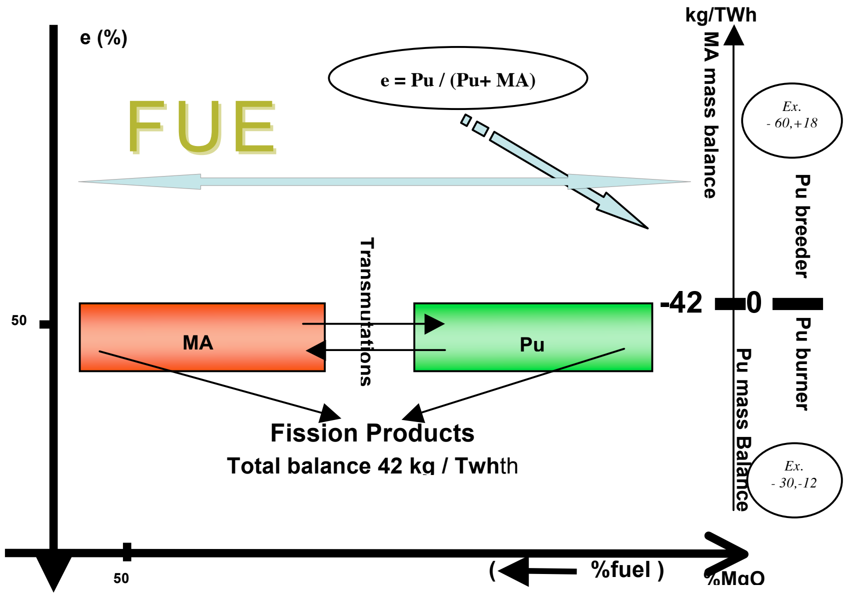


Fig. 5.24 - Additional MAs transmutation performances axis in the first quadrant of the A-BAQUS worksheet.

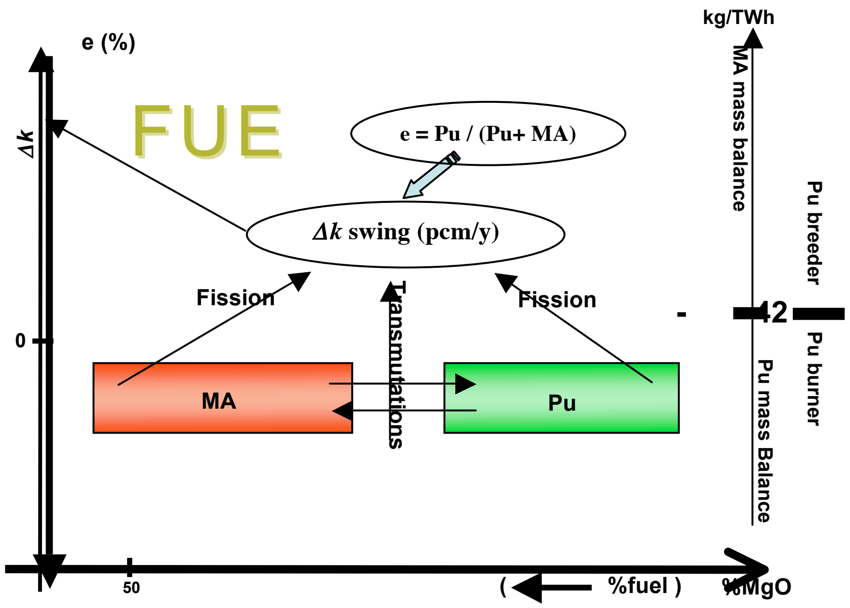


Fig. 5.25 - Additional criticality swing axis in the first quadrant of the A-BAQUS worksheet.

The fourth quadrant of the A-BAQUS worksheet (Figure 5.26) relates the fuel fraction in the pellet to the core thermal power. It is to be noticed indeed that, once the fuel pin and coolant designs are fixed, the higher the inert matrix fraction (thus the lower the fuel fraction), the more pins are needed to re-establish criticality (fuel enrichment being equal) increasing the core size and, in turn, increasing the core power.

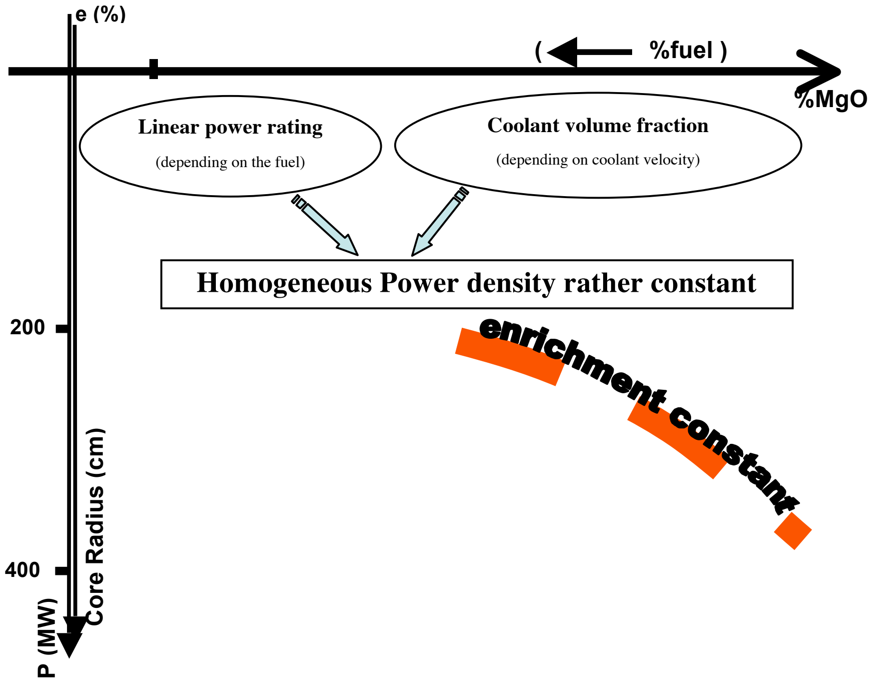


Fig. 5.26 - Fourth quadrant of the A-BAQUS worksheet.

The last (third) quadrant of the A-BAQUS worksheet relates the core power to the proton current to be provided by the accelerator module. It is known indeed that to maintain a constant power level during operation, a neutron source (by spallation from an in-core target) has to be provided, proportional to the aimed flux level in the system (thus to the core power P) and to the average number of neutrons per fission ν, and inversely proportional to both the fission Q-value and the average number of neutrons emitted by spallation per incident proton S. In order to relate the neutron source to the proton current, also the effective multiplication of the system, M_{eff}, and the specific multiplication of the neutron source, M_s, are to be accounted for. The overall law to determine the proton current i can be therefore expressed as

$$i = \frac{P\nu}{SQ}\frac{1-k_{\rm eff}}{\phi^* k_{\rm eff}} \tag{5.34}$$

where ϕ^* is the ratio between the source multiplication and the effective multiplication of the system:

$$\phi^* = \frac{M_s}{M_{\rm eff}} = \frac{\dfrac{k_{\rm s}}{1-k_{\rm s}}}{\dfrac{k_{\rm eff}}{1-k_{\rm eff}}}.$$

Finally, since the criticality of the system may evolve during operation, the accelerator is required to provide a proton current range to keep the power constant along the cycle, as shown in Figure 5.27.

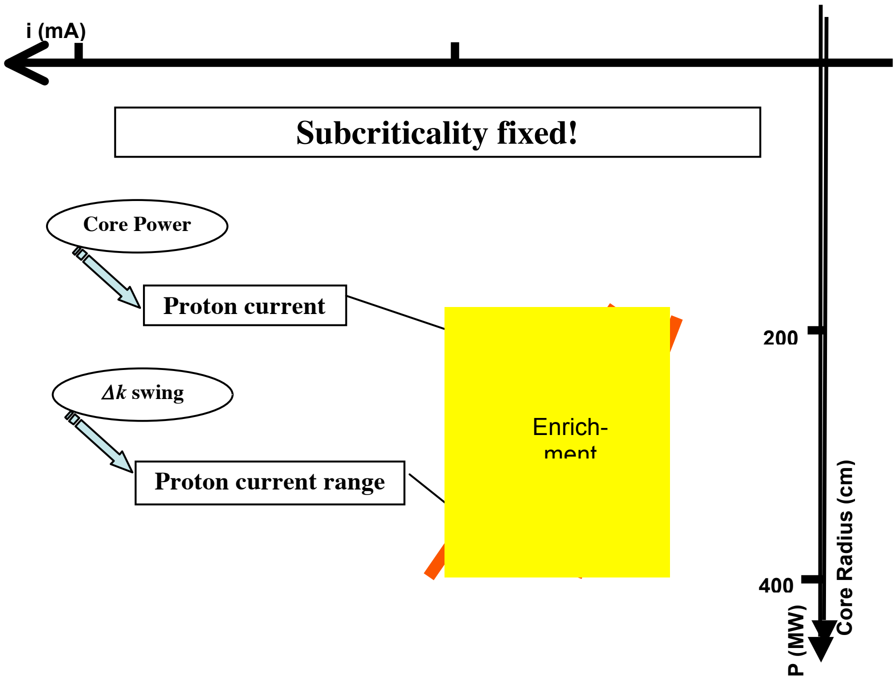


Fig. 5.27 - Third quadrant of the A-BAQUS worksheet.

Case study: EFIT

In order to design the EFIT ADS system, the A-BAQUS worksheet has been used to conceive the optimal configuration according to the aimed goal. Among the possible core optimizations (Δk swing = 0, high MAs transmutation rate, etc.), the 42-0 approach has been chosen as the leading criterion because of the general goal of "burning MAs at best" highlighted for the whole project (the zero-net Pu balance was also found consistent with the choice of a U-free fuel).

The technological constraints pointed out for the EFIT system are:

- 50% minimum matrix volumetric fraction (VF) to ensure the thermal conductivity of a CERCER $(Pu,MA)O_{2-x}$-MgO (thus U-free) fuel within the pellet;
- T_0 = 1650 °C for preventing inert matrix melting/disintegration (corresponding to a q'_{max} < 1800-200 W/cm, depending on the matrix VF);
- T_c = 550 °C for limiting corrosion;
- v < 1 m/s for limiting erosion effects.

An acceptable value of the coolant outlet temperature was also set to 480 °C.

Starting from the design of an elementary cell respecting all the technological constraints listed above, a set of preliminary calculations have been carried out to provide the information necessary to draw the respective curves on the A-BAQUS worksheet for different core optimization strategies, as shown in Figure 5.28.

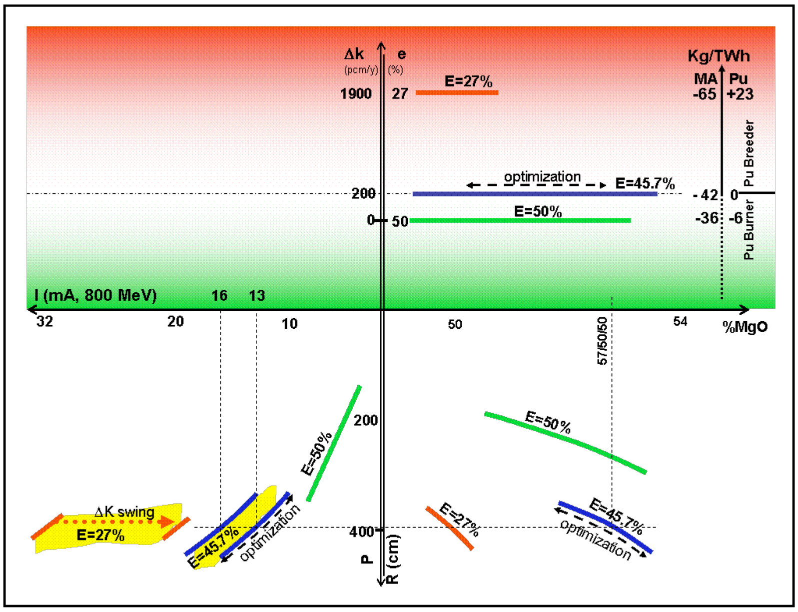


Fig. 5.28 - The A-BAQUS worksheet showing some curves correlating the core parameters under different optimization strategies.

The core design process, targeting also a maximization of the power density to keep the core dimensions (because of seismic risk) and a power/FA distribution flattening for improving the thermal yield of the plant (aiming at a costs minimization for the produced TWh) then led to the core parameters listed in Table 5.10.

TABLE 5.10 - EFIT main core parameters

Parameter	Reference value
Fuel pellet (solid) radius	3.55/3.55/4.00 mm
Matrix volumetric fraction	57/50/50 %
Gap thickness	0.16 mm
Clad thickness	0.60 mm
Fuel pin radius	4.31/4.31/4.76 mm
Pins lattice pitch (hexagonal)	13.63/13.63/13.54 mm
Active height	90 cm
Coolant velocity	1.00 m/s
k_{eff} (BoC)	0.97400
Δk swing (BoC - EoC)	500 pcm
Thermal power	400 MW
Proton current at BoC	13.2 mA

Since the fuel enrichment must be constant to guarantee the aimed MAs transmutation performances, the core flattening task has been pursued by segmenting the core into 3 zones with different fuel volumetric fractions. In order to simplify the construction of different pins, in the inner zone the matrix volumetric fraction has been increased (with respect to the values in the intermediate and outer fuel), while the outer zone relies on enlarged pins (as far as possible), as sketched out in Figure 5.29.

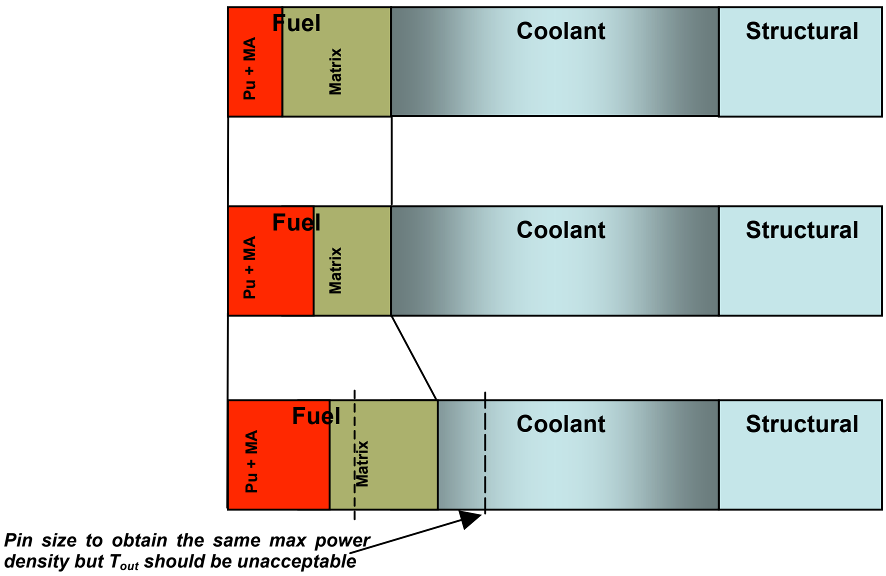


Fig. 5.29 - pellet (fuel and matrix), coolant and structural volumetric fractions in the different EFIT elementary cells.

A general view of the final EFIT core, according to the 42-0 approach, is shown in Figure 5.30.

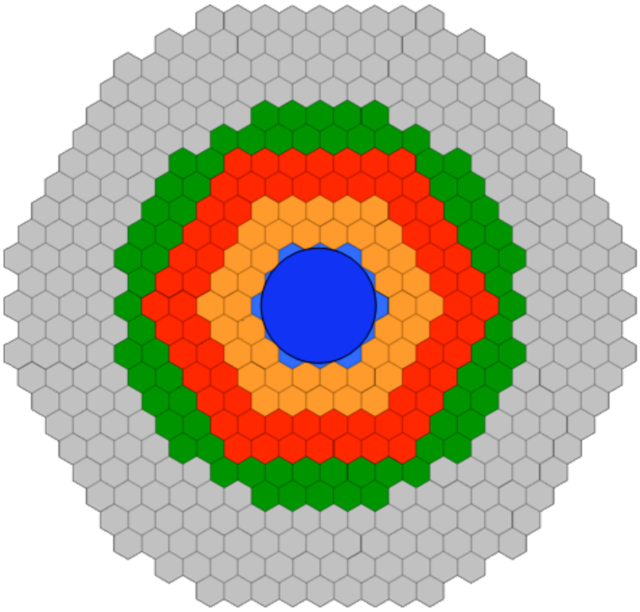


Fig. 5.30 - Final EFIT core scheme: orange, red and green and gray positions refer to inner, intermediate and outer FAs, grey positions represent the dummy elements and the blue zone hosts the spallation target.

6.b (iii) Adiabatic reactors

Aiming at designing adiabatic reactors, to ensure the sustainability of nuclear energy through the closure of the fuel cycle within the reactor itself, it is fundamental to clearly point out the parameters univocally defining the goal, borrowing the same approach implemented in the EFIT design (see section 5.6.b (ii)).

In order to design an adiabatic reactor, as a first step the equilibrium isotopic composition of the fuel must be fixed. This constraint in turn determines the intrinsic reactivity of the fuel: hence, the core designer is not able to design nuclear reactors to achieve an aimed power by setting the core size, and consequently adjusting criticality by tuning the fissile content in the fuel (section 5.6.b (i)); he must rather set up a critical arrangement for the given fuel.

According then to the thermal/hydraulic feasibility of the resulting core, and exploiting its viability, the system power will be univocally determined. This acts as - *si parva licet* - a “Copernican” revolution in the way of conceiving reactors, reversing the mental approach of subordinating the core design to its power: the whole design will be based on the fuel enrichment, fixed for the adiabaticity of the system; it will be possible then to tune the power by iteratively adjusting the elementary fuel cell and the corresponding fuel vector acting on the fuel volume fraction. A logical scheme for the design of an adiabatic core, according to this new paradigm, is shown in Figure 5.31.

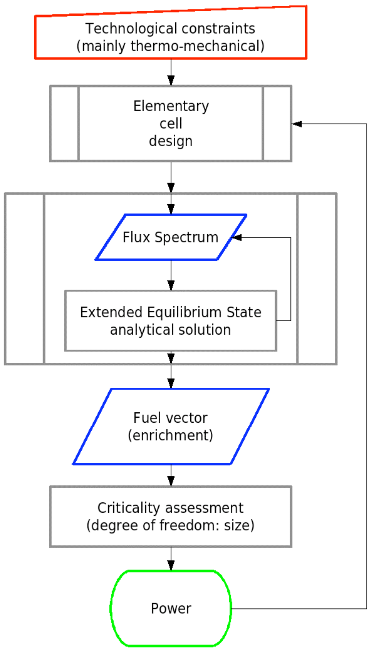


Fig. 5.31 - Logical flowchart for adiabatic reactors design according to the new paradigm.

The starting point for the whole process is the definition of the equilibrium vector. In order to retrieve the volume fractions of the materials in the elementary fuel cell (which determine the neutrons spectrum), a preliminary dimensioning of the fuel pin and coolant channel, *i.e.,* both the pin radius and lattice pitch, is needed. As described in the previous section (5.6.b (i)), it is possible indeed to determine those parameters *a priori*, by investigating the thermal/hydraulic consistency of the system according to the technological constraints represented by the allowable maximum temperatures for the coolant, the clad and the fuel as well as the maximum allowable coolant velocity and pressure drops through the core.

Once the fuel vector has been determined, whether its reactivity (*i.e.,* the k_∞ of the elementary cell) is enough higher than 1, the number of pins to be arranged in the core to get the criticality of the system is univocally determined, balancing the material buckling with the geometrical one. The number of pins in turn defines the corresponding core power. According to this reverted scheme, the dependency among the core parameters can be represented by the scheme of Figure 5.32.

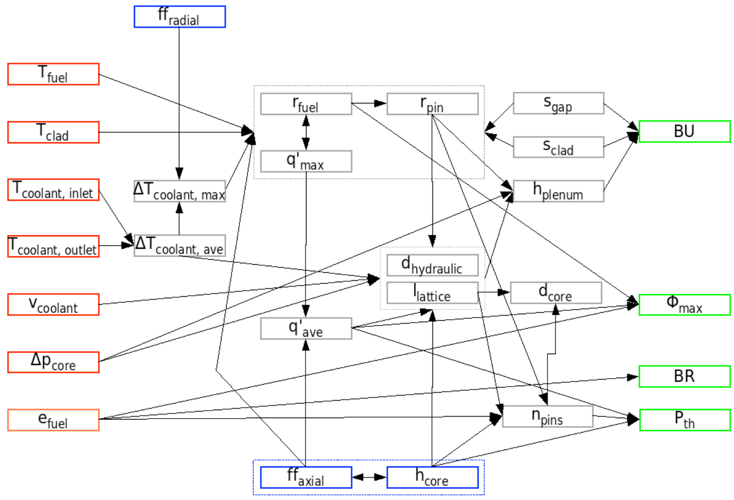


Fig. 5.32 - General scheme of adiabatic core parameters dependences in the new paradigm.

Hence, in an adiabatic core the dimensioning of the elementary cell uneqivocally determines a core power. The matching between the aimed power and the criticality of the system can be set by acting on the fuel volume fraction, thus re-defining from scratches the elementary cell in an iterative process (since the latter affects the neutrons spectrum).

6.c Design diagnostics and post-process feedbacks

As described in the previous section, according to the technological constraints, the core design is carried out by exploiting the mutual relationships among the core properties in order to outline the most exhaustive working point with respect to the aimed performances. In general,

- at first, the design constraints are translated into a set of viability ranges for the directly implied parameters;
- hence, axial and radial form factors are guessed (or inferred by previous analyses), corresponding to an initial hypothesis on the reactor shape; and
- the remaining equations are then put together and solved, taking also into account the design goals, providing the complete set of core parameters.

After the final core configuration has been assessed, a finalization phase follows, performing detailed calculations in order to retrieve actual estimates of the core performances. Besides a careful analysis to check the consistency of both system criticality and core temperatures to the design assumptions, further information must be retrieved, to be used as feedback information for the whole design process as well as to infer the core management strategies and anti-reactivity requirements.

6.c (i) Overall BU performances

The first feedback information comes from a detailed neutronics calculation accounting for the actual FAs refueling and reshuffling in the exact *n*-batches strategy to check the consistency of the 1-batch approximation preliminary assumed for core design.

According to this more detailed evaluation of fuel BU performances, both the actual criticality swing and power distribution evolution in the core (as a matter of fact, a segmentation of the core into zones with different fissile content – the most common solution to achieve power/FA distribution flattening – implies differently enriched FAs breed unevenly, altering the total power contributions redistribution) during the cycle can be checked.

This detailed analysis allows the determination of whether fuel swelling due to gaseous FPs does not overcome the designed in-clad void space, originating excessive PCMI, as well as determining that the limit for the maximum DpA on cladding is not exceeded.

According to these results, the allowed in-pile residence time and fuel management strategy can be fixed.

6.c (ii) Sizing and placement of control systems

A last analysis must be performed to check whether the supposed control, compensation and regulation systems are actually able to provide the required anti-reactivity for safe shutdown and cold arrest, as well as the anti-reactivity for criticality swing compensation and regulation during the cycle.

The results of this detailed analysis also provide useful information in order to resize or reposition the regulation/compensation and shutdown systems (taking also into account the required redundancy and differentiation) if needed.

6.d Reactivity coefficients

The evaluation of the reactivity coefficients for the system is the last step in core design. A complete list of parameters such as

- the coolant void reactivity worth,
- the Doppler coefficient,
- dimension coefficients and
- density coefficients

must be computed to provide the required information for kinetic and dynamic analyses of the system: the viability of a core configuration is assessed indeed after a complete safety analysis concerning both operative and incidental transients.

The required reactivity coefficients are evaluated simulating perturbed configurations where each parameter is singularly changed, and evaluating the criticality change for the system.

6.d (i) Lead void reactivity

The evaluation of the lead void reactivity coefficient is performed assuming all the coolant in the active zone is removed. Unlike SFRs, the lead boiling scenario can be assumed as unreal (the boiling temperature for lead being 1749 °C, far from common reactor coolant operating temperatures, vs. 883 °C boiling temperature for sodium): according to this, the complete voiding computation hypothesis is kept for coherence with sodium cooled reactors rather than for realistic accidental scenarios, even assuming large coolant losses (as for Loss Of Coolant Accident – LOCA, mitigated by the pool-type plant design) or strong injections of steam in the core following a massive Steam Generators Tubes Rupture (SGTR).

Nevertheless, the evaluation of the reactivity insertion due to the complete voiding of the cooling channels leads, for present LFR designs, to the typical values reported in Table 5.11.

TABLE 5.11 - Typical void reactivity coefficient of present LFR designs

System	Reactivity coefficient
ELSY	+4000 pcm
ENHS	+2700 pcm
EFIT	+6400 pcm
MYRRHA	-2300 pcm

It is worth noticing that different computation hypotheses are taken into account, referring to more realistic loss of coolant conditions. For instance, interesting results are obtained assuming that the core together with the upper and/or the radial reflectors is voided. Under such hypothesis, due to the high reflective power of lead, the coolant void coefficient is greatly reduced, even up to a sign change, as shown in Table 5.12 for ELSY.

TABLE 5.12 - Void reactivity coefficient of ELSY according to different voiding scenarios

Scenario	Reactivity coefficient
Active zone	+4042 pcm
Active zone and upper reflector	-1232 pcm
Active zone, upper and radial reflectors	-5251 pcm

It is worth noticing that, although negative void reactivity coefficient is not necessary for the safety of lead or LBE cooled fast reactors, in any case it is possible to re-conceive the core design to feature such a feedback. The main approaches for turning the positive void coefficient negative rely:

- on an enhancement of the neutrons leakage probability, for instance by
  - reducing the fuel length,
  - incorporating neutron absorbers in the core boundary,
  - using a gas-lift pump – that is, introducing gas bubbles throughout the coolant in the core and fission gas plenum regions,
  - incorporating neutron streaming channels in, and adjacent to the core;
- on the introduction into the core of materials having enhanced absorption cross-section at high energy;
- on the introduction into the core of materials that will keep the neutrons spectrum softer in case of coolant voiding.

6.d (ii) Doppler effect

The Doppler effect, acting as a self-shielding reduction because of absorption resonances broadening, is a main issue in reactor dynamics. Its effect on reactivity, behaving almost logarithmically as a function of the fuel temperature, is usually expressed by evaluating the Doppler coefficient α, defined as

$$\frac{\mathrm{d}k}{\mathrm{d}T_f} = \frac{\alpha}{T_f}$$

The Doppler coefficient is usually inferred by two criticality calculations on systems identical but for the fuel temperature. For ELSY (MOX fuel), a typical value of the Doppler coefficient results -700 pcm.

6.d (iii) Dimension and density reactivity coefficients

The reactivity variations of a system due to either dimensional or density perturbations provide useful information for transient analysis. The main dimension and density reactivity coefficients are expressed as

$$\frac{\partial \, \delta k / k}{\partial \, \delta p / p} \tag{5.34}$$

where p represents the perturbed parameter and δp the corresponding elementary perturbation.

A typical set of elementary perturbations, singly introduced to modify the reference system, lists:

- a radial extension of the core by scaling all radial dimensions, with nominal densities;
- an axial extension of the core by scaling all axial dimensions, with nominal densities (thus introducing some "slab" portion of core);
- a relative extraction of the CRs from their operative position;
- a reduction of the coolant density in the active zone;
- a reduction of the coolant density in the whole system;
- a reduction of the fuel density;
- a reduction of the steel density;
- a reduction of the absorbers density.

Some other coefficient may be added to the list above, to complete the set of information regarding the core neutronics, such as:

- an increase of the Pu enrichments in the core;
- a reduction of the U density in the fuel (maintaining the Pu density unchanged);
- a reduction of the Pu density in the fuel (maintaining the U density unchanged).

Once the set of elementary perturbations has been pointed out, some relative variations have to be assumed for each parameter in order to define the perturbed configuration to be simulated. A typical computational scheme (showing the values assumed for each elementary perturbation and the corresponding effect on criticality for ELSY) is resumed in Table 5.13.

TABLE 5.13 - ELSY computational scheme for dimension and density reactivity coefficients evaluation and corresponding reactivity effect

Perturbation	**Variation**	**Δk_{eff}**
Radial extension of the core	R_{core} +2.5%	+239
Axial extension of the core	H_{core} +5%	+842
Partial extraction of absorbers from the core	L_{ins} -1 cm	+36
Reduction of coolant density in the core	ρ_{cool}^{core} -5%	+161
Reduction of coolant density in the whole system	ρ_{cool}^{sys} -5%	-22
Reduction of fuel density	ρ_{fuel} -5%	-1614
Reduction of steel density	ρ_{steel} -5%	+170
Reduction of absorbers density	ρ_{abs} -5%	+13
Increase of Pu enrichments	E_{Pu} +1 pt	+3507
Increase of U density in the fuel	ρ_U +5%	-1190
Increase of Pu density in the fuel	ρ_{Pu} +5%	+2734

By combining the computed criticality change to the relative perturbation of the corresponding parameter, the aimed dimension and density reactivity coefficients can be finally retrieved.

6.d (iv) Feedback reactivity coefficients

The dimension and density reactivity coefficients introduced in the previous subsection provide the elementary information to compute the feedback reactivity coefficients used for actual system transient analysis. As a matter of fact, every transient the system undergoes is the result of a complex combination of a multitude of single effects: for instance, in case of positive Transient Of Power (TOP), every material in the core increases its temperature so that, besides the most immediate Doppler and density effects, also geometrical effects must be accounted because of the dilation of the whole system.

Therefore, in order to provide an unique combined reactivity coefficient, all the involved effects, examined in the previous step, must be related to a common parameter driving all the elementary perturbations. The most suitable parameter, which can also be identified the cause of all perturbations, is temperature. The aimed feedback reactivity coefficient will be therefore expressed as

$$\frac{\partial \, {}^{\delta k}\!/_{k}}{\partial T} = \sum_i \frac{\partial \, {}^{\delta k}\!/_{k}}{\partial \, {}^{\delta p_i}\!/_{p_i}} \frac{\partial \, {}^{\delta p_i}\!/_{p_i}}{\partial T} \tag{5.35}$$

where the sum is extended over all the elementary contributions participating to the effect under investigation.

In general, also the feedback reactivity coefficients are separated to provide a more flexible input capability to security analysis tools. The most common feedback reactivity coefficients are therefore related to the diagrid-driven radial dilation of the core, and to the axial dilation of the latter.

To what concerns the diagrid-driven dilation feedback reactivity coefficient, the following expression relating the single reactivity coefficients is adopted.

$$\begin{aligned} \left.\frac{\partial \, {}^{\delta k}\!/_{k}}{\partial T}\right|_{\text{diagrid}} = \frac{\partial \frac{\delta R_{core}}{R_{core}}}{\partial T} \left\{ \frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta R_{core}}{R_{core}}} + \frac{\partial \frac{\delta \rho}{\rho}}{\partial \frac{\delta R_{core}}{R_{core}}} \left[\frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta \rho_{fuel}}{\rho_{fuel}}} + \right.\right. \\ \left.\left. + \frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta \rho_{steel}}{\rho_{steel}}} + \frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta \rho_{abs}}{\rho_{abs}}} + \left(1 - \frac{\partial \frac{\delta \rho_{cool}}{\rho_{cool}}}{\partial \frac{\delta \rho}{\rho}} \right) \frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta \rho_{cool}}{\rho_{cool}}} \right] \right\} \end{aligned} \tag{5.36}$$

A similar expression holds also in the case of axial dilation of the system:

$$\left.\frac{\partial \,{}^{\delta k}\!/_{k}}{\partial T}\right|_{axial} = \frac{\partial \frac{\delta H_{core}}{H_{core}}}{\partial T}\left\{\frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta H_{core}}{H_{core}}} + \frac{\partial \frac{\delta \rho}{\rho}}{\partial \frac{\delta H_{core}}{H_{core}}}\left[\frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta \rho_{fuel}}{\rho_{fuel}}} + \right.\right.$$

$$\left.\left. + \frac{\partial \frac{\delta k}{k}}{\partial \frac{\delta \rho_{steel}}{\rho_{steel}}}\right]\right\} + \frac{\partial \frac{\delta k}{k}}{\partial l_{CRs\,insertion}} \frac{\partial l_{CRs\,insertion}}{\partial T} \tag{5.37}$$

7. Reactor System

All primary system configurations proposed so far for a LFR are of the pool type. This is the obvious result of the cost and technical difficulties associated with a loop type configuration. In fact at present good design practice is to limit the lead speed to 2 m/s to reduce both pressure loss and erosion of structural material and this would result in large diameter heavy tubes for a loop type reactor.

Several configurations have been proposed for the primary system ranging from the natural circulation (SSTAR) solution, the enhanced circulation solution using gas injection (XT-ADS) in the riser, and the solution of forced circulation (ELSY).

Natural circulation is convenient for simplification of small reactor (tens of MWe) whereas forced circulation is necessary for compactness of large reactors (hundreds of MWe). Mechanical pumps are generally proposed because of the low efficiency of electromagnetic pumps in lead.

Because of the low allowed lead speed, the primary flow path must be simple and as short as possible to reduce the mass of lead to guarantee a successful mechanical behavior under seismic loads for which a reduced vessel length is an additional need.

7.a Reactor Vessel and Safety Vessel

It is a classical approach in case of a liquid metal cooled reactor to have a Reactor Vessel which contains the primary system surrounded by a safety vessel that collect potential leakage of coolant from the Reactor Vessel.

The Reactor Vessel is in general shaped as a cylinder with a hemispherical bottom and a flat roof. The lead level is kept below the roof to accommodate the thermal gradient between the vessel in contact with lead and the colder roof. The

Reactor Vessel can be supported directly by the roof as in SSTAR or by a connection below the roof to a conical shell as in ELSY. The object of the ELSY solution is to separate the mechanical load due to the lead weight from the thermal gradient of the connection to the roof.

The roof is a thick plate with penetrations for the components and the above core structures which are laded on it.

The safety vessel can be conceived as an additional steel vessel surrounding the reactor vessel or can be integrated in the reactor pit as a liner of the concrete walls. In the latter case the safety vessel is protected, reactor side, by an insulating layer and is kept cold by a Reactor Concrete Cooling System (RCCS) consisting of water pipes located inside the reactor pit concrete.

Reactor Vessel Air Cooling (RVACS) system pipes can be located outside the safety vessel in the first case and between the two vessels in the second case.

The volume above the lead free level is filled with inert gas.

7.b Reactor Internal Structures

The cylindrical inner vessel configuration is the classical configuration adopted LFRs in natural circulation because of its simplicity and reduced pressure loss. This configuration is characterized by a core, located centerline in the bottom part of the Reactor Vessel and its upper structures surrounded by a cylindrical structure which contains inside the hot lead of the riser and, with the Reactor Vessel, delimits an annular volume of cold lead where the main component are located, namely the steam generators, the pumps, the purification units the dip coolers of the DHR systems (Figure 5.33). The differential weight of the lead inside the riser and the lead in the riser produce the driving head necessary for the natural circulation of lead of the primary system. SGs can be freely installed inside the cold collector, with the inconvenience of a thermal stratification in the cold collector and in the Reactor Vessel or hydraulically connected with more complicate reactors internals and additional difficulties for component replacement.

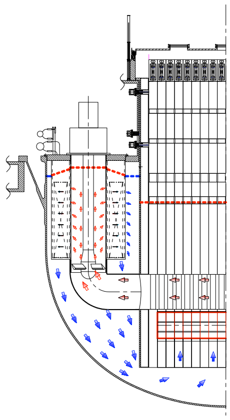


Fig. 5.33 - Detail of the ELSY primary system arrangement and coolant flow path.

In case of forced circulation, the hydraulic connection between the SG and the riser is definitely necessary.

A particular innovative solution has been identified in ELSY to adopt an inner vessel of perfect cylindrical shape while ducts are mechanically connected to the SGs to be fed. The cylindrical inner vessel, as usual, constitutes the lateral restraint of the core, but differently from previous solutions is not connected to the core support plate which can be avoided thanks to more advanced solutions. The core support plate constitutes in general a critical component submitted to fast neutron flux, difficult to replace and with difficulty/impossibility of ISI and repair. A simple cylindrical inner vessel can be supported in the upper part by the roof with a releasable connection for its replacement in case of need.

A peculiar load to be considered for the seismic design of the internals of a LFR is the load associated to lead sloshing that can be only partially mitigated by the adoption of seismic isolators. In fact seismic isolators of the reactor building can drastically reduce the acceleration of the reactor structures but also lower the frequencies and move them closer to the frequencies typical of the sloshing phenomena.

To be removable, the internals can be hung from and supported by the reactor roof, a metallic plate welded to the reactor vessel. The reactor roof with its sealed penetration for the components together with the reactor vessel constitute the primary containment.

7.c Steam Generator

Several types of SG have been proposed for LFRs, the most common being the helical tube SG for which a deep experience exists for SFR applications. An innovative SG has been introduced instead in the ELSY project looking to several advantages in term of reactor cost, safety and reactor operability and simplicity of the lead flow path.

This innovative SG is composed of a stack of spiral-wound tube bundle (Figure 5.34) arranged in the bottom-closed, annular space formed by a vertical outer and an inner shells. 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 one above the other and equally spaced.

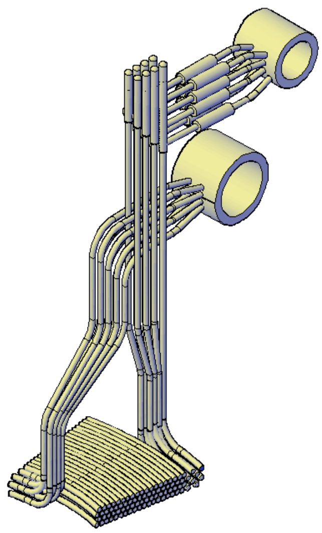


Fig. 5.34 - SG with a spiral-wound tube bundle.

The coolant flows radially through the perforated inner shell and, past the tube spirals, through the outer shell. This scheme is thermally almost equivalent to a pure counter-current scheme, because the feed water in the tube circulates from the outer spiral to the inner spiral, while the primary coolant flows in the opposite direction from the inner shell to the outer shell of the SG. There is no window as 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 leaking Reactor Vessel, in fact the shell perforations extend below the accidental coolant free level and ensure adequate flowrate for core cooling. As a by-product, the SG unit can be positioned at a higher level in the downcomer and the RV shortened, accordingly.

The suction pipe is an integral part of the SG bottom structure and extends outside the SG circular orthogonal projection to match the contour of the port cut out in the wall of the Cylindrical Inner Vessel. The horizontal duct between the SG and the Inner Vessel normally constitutes a major obstacle for the replacement of the component to which is connected, namely the SG or the Inner Vessel because of its interference with the smaller penetration through the reactor roof. Feeding the SG from the bottom offers the additional advantage of providing a procedure to extract the SG from the Reactor vessel provided that the two geometrical condition of Figure 5.35 are satisfied.

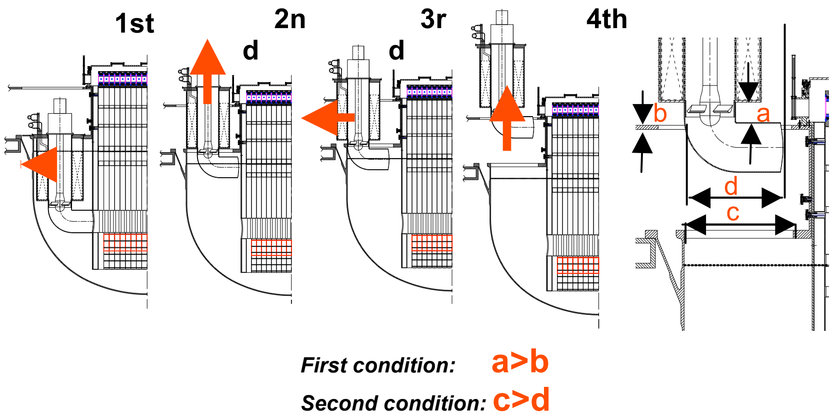


Fig. 5.35 - Geometrical conditions for the SG replacement with the connecting duct

The first small displacement, radial, aims at disengaging the SG horizontal duct from the Inner Vessel taking profit of the clearance between the SG and its penetration through the reactor roof. The second displacement, vertical, brings the horizontal duct nearly in contact with the lower surface of the reactor roof. The third displacement, radial, brings the horizontal duct inside the orthogonal projection of the roof penetration. The fourth displacement, vertical, allows the complete extraction of the SG from the Reactor Vessel.

With removable SG's and PP's, also the Cylindrical Inner Vessel can be designed as a removable Unit, and eventually the design goal of all removable Internals becomes feasible.

It should be noted that the reason in favour of the helical-tube SG, with respect to other conventional SG concepts, has always been that it copes better with high thermal loading, in spite of higher cost.

The rationale of the spiral-tube SG vs the helical-tube GV can be stated as follows:

- tolerant to thermal loading as the helical-tube GV;
- predictable lower cost because the tube spirals are easier to assemble and require simpler supports;
- adequately fed also in case of coolant free level drop further to the Reactor Vessel leakage accident;
- less space required (tube bundle volume reduced of a factor two owing to the simpler tube support system) and shell-side pressure loss reduced by about factor two (less tubes to flow through).

The installation of SGs inside the reactor vessel is major challenge of a LFR design which includes the need for a sensitive and reliable leak detection system and a highly reliable depressurization and isolation system.

In ELSY careful attention has been given to the issue of mitigating the consequences of the Steam Generator Tube Rupture (SGTR) accident to reduce the risk of pressurization of the primary boundary; to this end, innovative provisions have been conceived which make the primary system more tolerant of the SGTR event.

The first provision is the elimination of the risk of failure of the water and steam collectors inside the primary boundary by installing them outside the reactor vessel. This approach aims to eliminate by design a potential initiator of a severe accident of low probability but potentially catastrophic consequences.

The second provision is the installation on each tube of a check valve close to the steam header and of a Venturi nozzle close to the feed water header.

The third provision aims at ensuring that the flow of any feedwater-steam-primary coolant mixture be re-directed upwards inside the SG, reducing by design the risk of propagation of large pressure waves across the reactor vessel. This occurs because the inner pressure surge itself promptly causes the closure of the normal radial coolant flow path. Lead overflow from the SG into the downcomer at reactor free level damps the pressure surge without risk of serious damage of the reactor internals[9].

The fourth provision is the installation on the reactor roof of pressure relieving ducts each with rupture discs, connecting the reactor cover gas plenum with the Above-Reactor Enclosure (ARE) to limit the pressure surge inside the reactor vessel.

7.d Primary coolant circulation

Small size reactors (e.g., SSTAR) can rely on lead natural draft which can be of the order of 1500 Pa for each meter of relative elevation between core and SG.

The use of air lift can deliver a draft of about 5000 Pa for each meter of the riser length and it can be a solution to shorten the reactor vessel of small size reactors in comparison to the use of the natural circulation.

Forced circulation with mechanical or electromagnetic pumps is necessary to deliver a head of 1-2 bar necessary to reduce the size of large power reactors.

At present electromagnetic pumps has been disregarded by all LFR designers, presumably because of their low efficiency.

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[9] Perforated companion inner and outer shells are placed close to inner and outer shell respectively, held apart to a few mm by spacers. The spacers are designed to collapse in case the inner companion shells are acted upon by a specified inner pressure. Thus, in case of inner pressure surge, the companion shells blow out against inner and outer shell respectively and since the holes of the corresponding perforations have been designed staggered and the bottom end of the annulus is closed, the mixture will flow upwards towards the cover gas plenum.

Mechanical pumps for LFRs are a suitable solution with high efficiency and great simplicity. A pump impeller few meter deep in lead can guarantee the required NPSH and consequently a short shaft is sufficient to connect the pump impeller to the pump motor located on the reactor roof. No supporting bearing in lead is necessary. In case of ELSY for additional compactness the shaft and the impeller of the pump are located in a free volume inside the spiral tube SG.

8. Decay Heat Removal System.

A LFR normally relies on the secondary system (the water-steam system, in the case of ELSY) to remove decay heat.

The water-steam system, however, is not a safety-grade system and additional more reliable safety-grade systems are necessary to meet the safety objectives.

A reliable system for decay heat removal is the Reactor Vessel Air Cooling System (RVACS).

Unfortunately, the RVACS by itself can be used only in small-size reactors, the reactor vessel outer surface of which is relatively large to enable the transfer of the generated reactor decay power.

For a large power reactor it is necessary to install additional loops equipped with coolers immersed in the primary coolant, a decay heat removal system called hereinafter the DRC (Direct Reactor Cooling) system.

The DRC loops, because of their greater complexity, will result in a lower reliability than the simple RVACS. Stringent safety and reliability requirements of the DRC system will be achieved by redundancy and diversification.

The DRC system is comprised of loops which can operate with stored water (W-DHR) or external air (A-DHR).

Additional cooling functions are also necessary to permanently cool the concrete of the reactor pit and to control the air temperature of the reactor pit itself during in-service inspection (ISI) of the reactor vessel.

8.a Reactor Vessel Air Cooling System

Different RVACS configurations have been proposed for LFR based on SFR experience. The RVACS system developed for ELSY consists basically of an annular pipe bundle of U-pipes arranged between the reactor vessel and the safety vessel in a nest type configuration with atmospheric air flowing pipe-side in natural or forced circulation (Figure 5.36). In spite of the improvements in this design relative to earlier concepts, even in ELSY, the performance is sufficient only in the long term (after about one month after shut down) and additional loops are needed for short-term decay heat removal.

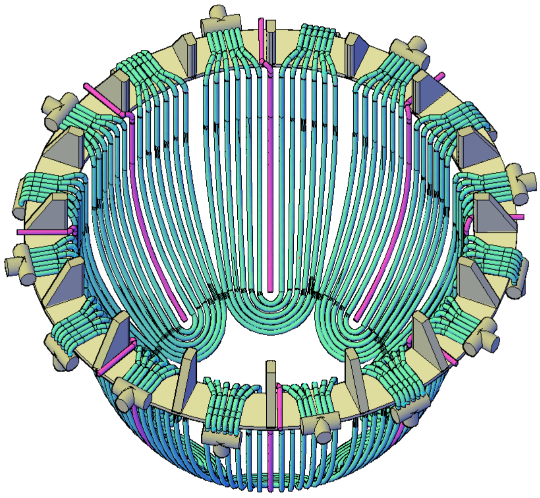

Fig. 5.36 - The nest configuration of the ELSY RVACS.

8.b Water loops and associated dip coolers

The fact that molten lead does not react violently with air or water gives the designer some freedom in the choice of the coolants to be used in the DHR loops, the use of air and water remaining the preferred approach.

A typical scheme of a Direct Reactor Cooling (DRC) system for LFR based on water, the W-DHR loops, with coolers immersed in the primary system, is presented in Figure 5.37.

Each W-DHR loop is made of a cooling water Storage Tank, a water-lead Dip Cooler, interconnecting piping, and steam vent piping to discharge steam to the atmosphere.

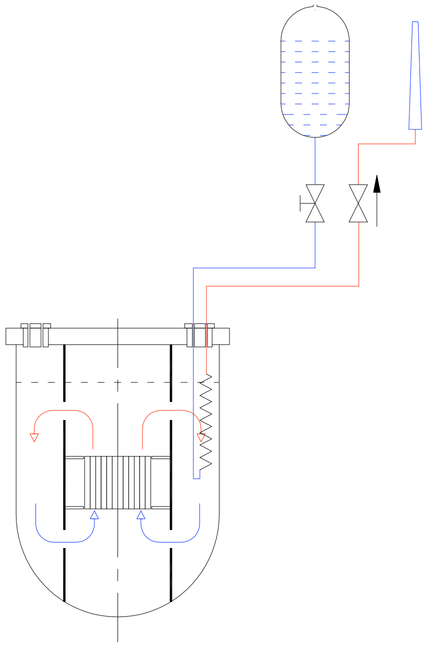

Fig. 5.37 - The DRC W-DHR (right-side) and WA-DHR loops, process scheme showing stored cooling water interconnection.

The Dip Cooler tube bundle is made of bayonet tubes (see Figure 5.38). The bayonet consists of three concentric tubes, the outer two of which have the bottom end sealed. Water evaporation or air heating takes place in the annulus between inner tube and the intermediate tube. The annulus between the outer tube and intermediate tube is filled with He gas at a pressure higher than the lead pressure at the bottom end of the bundle. All annuli are interconnected to form a common He gas plenum, the pressure of which is continuously monitored. A leak from either wall of any of the outer tubes, is promptly detected because of depressurization of the common gas plenum.

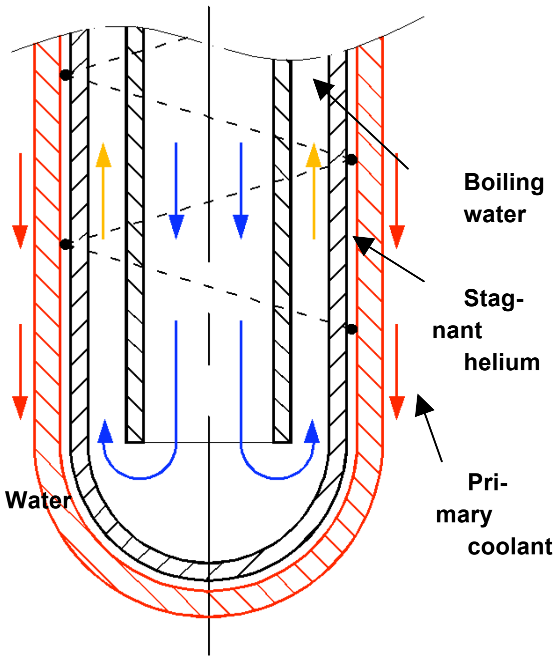

Fig. 5.38 - Bayonet tubes of the DHR dip coolers.

The proposed bayonet RC dip coolers are different with respect to classical bayonets, which consist each of only a pair of concentric tubes. The two outer tubes do not constitute a double walled tube, but are mechanically and, thermally decoupled. This configuration allows to localize the most part of the thermal gradient between lead and boiling water across the gas layer, avoiding both risk of lead freezing and excessive thermal stresses across the tube walls during DHR steady state operation and transients.

The typical outer diameter of the outer tube is about 25 mm.

8.c Air loops and associated dip coolers

A diversified system operating in air, an A-DHR loop is made of an inlet air duct, an air-lead dip cooler and an outlet air duct. The inlet air duct is equipped with an electric fan supplied by batteries. Isolation valves are installed in the inlet air and outlet ducts.

The dip cooler of an A-DHR loop can be based on the same principle as the W-DHR cooler, but to allow a sufficient air flow rate with the air side pressure loss compatible with air natural circulation or with low power fans, the resulting minimum outer diameter of the outer tube is on the order of 150 mm. The resulting dip cooler is of relatively large cross-section and impacts the reactor vessel diameter, but has the advantage of constituting a passive solution which can operate in the long term when the storage water of the A-DHR loops is exhausted.

9. Nuclear Island

The following considerations on LFRs are based on the hypotheses of a central reprocessing and fuel fabrication plant physically separated from the reactor. This is applicable to both the small reactors (SSTAR type) and large reactors (ELSY type).

As regards to the spent fuel reprocessing and fabrication of fresh fuel, the situation of the LFR is similar to that of the SFR.

A significant difference among the two LFR systems is that SSTAR foresees the supply and replacement of the entire core, whereas ELSY foresees quite standard operational practices with periodic access to the core for fuel handling and partial replacement of the core.

It should be noted that the genesis for the SSTAR concept was the idea of developing a reactor that was, by design, low in proliferation risk and therefore deployable virtually anywhere in the world. The objectives resulting from this goal included factory fabrication (and fuelling); transportability of the reactor system to the site and installation without the requirement for handling fresh fuel or for developing a fuel supply infrastructure; ultra-long core life to enable long-term operation without refuelling; and robustness and simplicity of design (e.g., reliance on natural convection flow for heat removal) to minimize operational complexity and maintenance requirements.

In the case of ELSY, considerable work has been carried out to define the overall plant layout. Figure 5.39 below provides an overview sketch of the current reference plant layout.

The reference design shown incorporated forced-draft cooling towers. A second option has also been studied based on natural-draft cooling towers.

The ELSY Reactor Building is a six story building, two stories of which are below ground level. It is of cylindrical shape. Its base plate, located below grade, rests on seismic supports and a single foundation slab. The lowest floor is the storage area for fresh and spent fuel assemblies.

With respect to spent fuel, it is possible either (i) to store all spent fuel inside the reactor building or (ii) to provide a limited storage capacity inside the reactor building (namely, sufficient storage for a single core) with additional capacity in an auxiliary dedicated building.

The reactor building is designed to withstand anticipated earthquake stresses and it is provided with double barrier containment. The outer containment barrier is made of reinforced concrete with a steel liner on the inner surface, and is designed to withstand the double-ended rupture of one main steam manifold.

The Above Reactor Enclosure (ARE) performs as the first containment barrier and contained work area whenever the vessel head is removed and in-vessel components and fuel assemblies are lifted from the reactor vessel by means of large and small cranes, respectively, both cranes being arranged in the ARE.

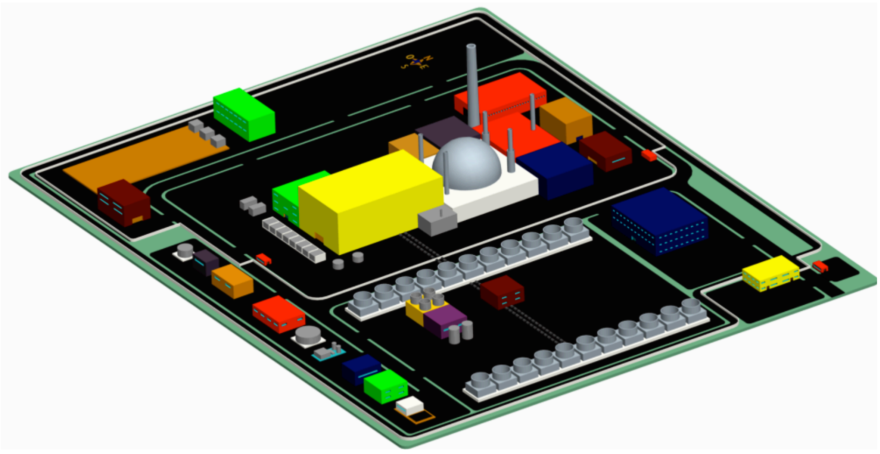

Fig. 5.39 - ELSY general layout.

Besides the Reactor Vessel, the Reactor Building houses water storage pools required to supply the safety-grade Direct Reactor Cooling System (DRC system) and the piping for the Reactor Vessel Air Cooling System (RVACS). Two additional water storage pools for the Secondary Loops Reactor Cooling System are located outside the Reactor Building at both sides of the steam tunnel. The three DHR systems are connected to four chimney stacks, allowing for the release of the RVACS hot air and the steam of the other systems.

The four chimneys are arranged symmetrically around the Reactor Building, one chimney stack in each quadrant.

The reactor building is supported by seismic isolation bearings to decouple the building from the ground, lengthening the period of the building and lowering the response for the structures.

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