Lead-cooled Fast Reactor (LFR) Risk and Safety Assessment White Paper

A. Alemberti, M.L. Frogheri, S. Hermsmeyer, L. Ammirabile V. Smirnov, M. Takahashi, C.F. Smith, Y. Wu, I.S. Hwang

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Abstract

This paper presents the application of the Integrated Safety Assessment Methodology (ISAM), developed by the Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG), to the Lead-Cooled Fast Reactor (LFR). In this effort, the LFR-provisional System Steering Committee (pSSC) determined that the information available on the reference systems for GIF activities was not sufficient for an accurate and complete application of the ISAM. As a consequence, the LFR-pSSC decided to include in the paper, to the extent possible, a summary of the existing common features of the three LFR reference systems (ELFR, BREST, SSTAR), and then present an application of ISAM to the ALFRED demonstrator since, for this system, a consistent set of information has been disclosed and is available for the application. After presenting a short overview of ISAM and a short summary of the three reference LFR systems, the ALFRED design is reported in more detail and then the results of the ISAM application are summarized.

1. Short overview of the ISAM assessment methodology

The RSWG has developed a methodology, called the Integrated Safety Assessment Methodology (ISAM), for use throughout the Gen IV technology development cycle. The ISAM consists of five distinct analytical tools (Ref. 1) which are intended to promote safety that is "built-in" rather than "added on" by influencing the direction of the concept and design development from the earliest stages. The ISAM tools are the following:

- Qualitative Safety Features Review (QSR)
- Phenomena Identification and Ranking Table (PIRT)
- Objective Provision Tree (OPT)
- Deterministic and Phenomenological Analyses (DPA)
- Probabilistic Safety Analysis (PSA)

Figure 1.1 shows the overall task flow of the ISAM and indicates which tools are intended for use in each phase of Generation IV system technology development.

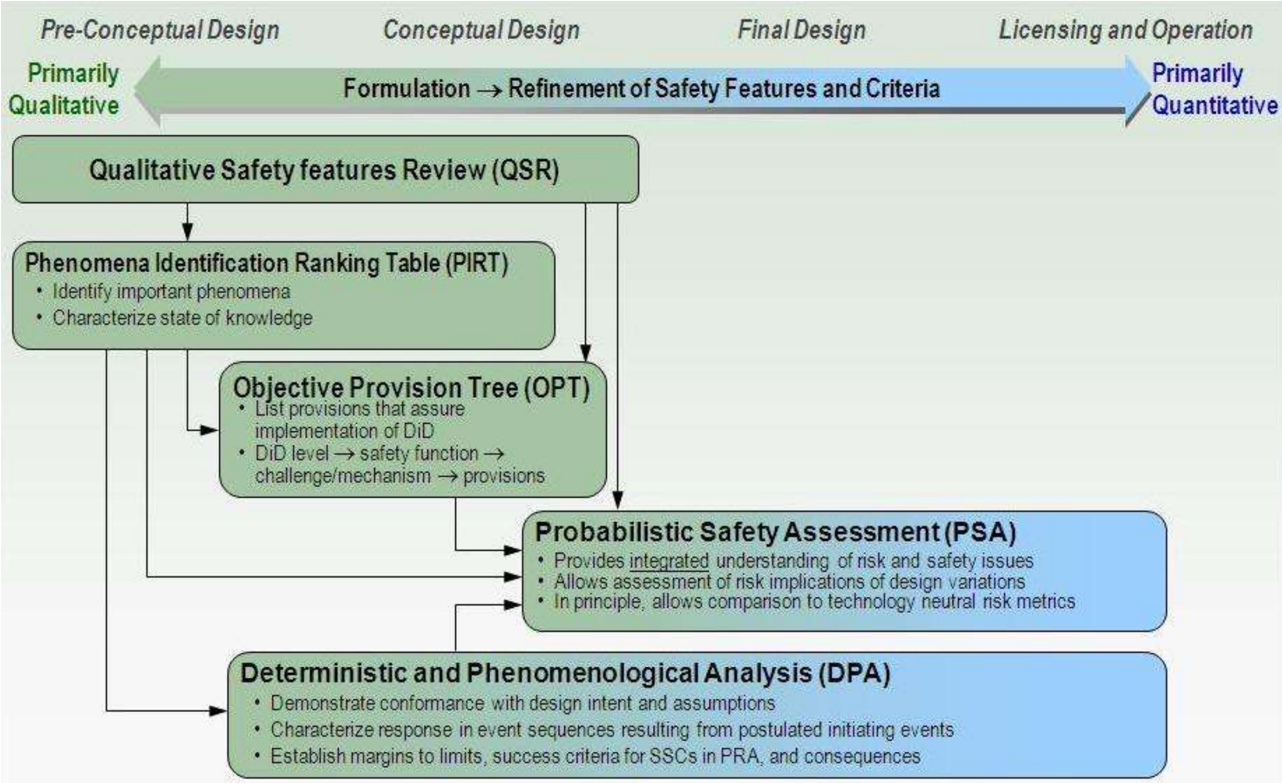

Figure 1.1: Proposed GIF Integrated Safety Assessment Methodology (ISAM) Task Flow

Each of the analysis tools that is part of the ISAM is briefly described below.

- Qualitative Safety Features Review (QSR)

The Qualitative Safety Features Review (QSR) is a new tool that provides a systematic means of ensuring and documenting that the evolving Gen IV system concept of design incorporates the desirable safety-related attributes and characteristics that are identified and discussed in the RSWG's first report entitled, "Basis for the Safety Approach for Design and Assessment of Generation IV Nuclear Systems", as well as in other references (e.g., the INPRO Safety methodology). Although this element of the ISAM is offered as an optional step, it is believed that the QSR provides a useful means of shaping designers' approaches to their work to help ensure that safety truly is "built-in, not added-onto" from the early phases of the design of Gen IV systems. Using a structured template to guide the process, concept and design developers are prompted to consider, for their respective systems, how the attributes of Defence in Depth (DiD), high safety reliability, minimization of sensitivity to human error, and other important safety characteristics might best be incorporated. The QSR also serves as a useful preparatory step for other elements of the ISAM by promoting a richer understanding of the developing design in terms of safety issues or vulnerabilities that will be analyzed in more depth in other analytical steps.

- Phenomena Identification and Ranking Table (PIRT)

The Phenomena Identification and Ranking Table (PIRT) is a technique that has been widely applied in both nuclear and non-nuclear applications. As applied to Gen IV nuclear systems, the PIRT is used to identify a spectrum of safety-related phenomena or scenarios that could affect those systems, and to rank those phenomena or scenarios on the basis of their importance (often related to their potential consequences), and the state of knowledge related to associated phenomena (i.e. sources and magnitudes of phenomenological uncertainties).

The method relies heavily on expert elicitation, but provides a discipline for identifying those issues that will undergo more rigorous analysis using the other tools that comprise the ISAM. As such, the PIRT forms an input to both the Objective Provision Tree (OPT) analyses, and the Probabilistic Safety Analysis (PSA). The PIRT is particularly helpful in defining the course of accident sequences, and defining safety system success criteria. The PIRT is essential in helping to identify areas in which additional research may be helpful to reduce uncertainties.

- Objective Provision Tree (OPT)

The Objective Provision Tree (OPT) is a relatively new analytical tool that is enjoying increasing use. The International Atomic Energy Agency (IAEA) has been a particularly influential developer and proponent of this analysis tool. The purpose of the OPT is to ensure and document the provision of essential "lines of protection" to ensure successful prevention, control or mitigation of phenomena that could potentially damage the nuclear system. There is a natural interface between the OPT and the PIRT in that the PIRT identifies phenomena and issues that could potentially be important to safety, and the OPT focuses on identifying design provisions intended to prevent, control, or mitigate the consequences of those phenomena.

- Deterministic and Phenomenological Analyses (DPA)

Classical deterministic and phenomenological analyses, including thermal-hydraulic analyses, computational fluid dynamics (CFD) analyses, reactor physics analyses, accident simulation, materials behavior models, structural analysis models, and other similar analysis tools collectively constitute a vital part of the overall Gen IV ISAM. These traditional deterministic analyses will be used as needed to understand a wide range of safety issues that guide concept and design development, and will form inputs into the PSA. These analyses typically involve the use of familiar deterministic safety analysis codes. It is anticipated that DPA will be used from the late portion of the pre-conceptual design phase through ultimate licensing and regulation of the Generation IV system.

- Probabilistic Safety Analysis (PSA)

Probabilistic Safety Analysis (PSA) is a widely accepted, integrative method that is rigorous, disciplined, and systematic, and therefore it forms the principal basis of the ISAM. PSA can only be meaningfully applied to a design that has reached a sufficient level of maturity and detail. Thus, PSA addresses licensing and regulatory concerns and is performed, and iterated with a beginning in the late pre-conceptual design phase, and continuing through to the final design stages. In fact, as the concept of the "living PSA" (one that is frequently updated to reflect changes in design, system configuration, and operating procedures) is becoming increasingly accepted, the RSWG advocates the idea of applying PSA at the earliest practical point in the design process, and continuing to use it as a key decision tool throughout the life of the plant or system. Although the other elements of the ISAM have significant value as stand-alone analysis methods, their value is enhanced by the fact that they serve as useful tools in helping to prepare for and to shape the PSA once the design has matured to a point where the PSA can be successfully applied.

It is intended that each tool be used to answer specific kinds of safety-related questions in differing degrees of detail, and at different stages of design maturity. As indicated in Ref. 1 it is envisioned that the ISAM and its tools will be used in three principal ways:

- Use throughout the concept development and design phases with insights derived from the ISAM serving to influence the course of the design evolution.

- Punctual implementation of selected elements of the methodology which are applied at various points throughout the design evolution to yield an objective understanding of risk contributors, safety margins, effectiveness of safety-related design provisions, sources and impacts of uncertainties, and other safety-related issues that are important to decision makers.

- An application in the late stages of design maturity to measure the level of safety and risk associated with a given design relative to safety objectives or licensing criteria.

2. Overview of Technology

The Generation IV Technology Roadmap [1], prepared by Generation IV International Forum (GIF) members, identified the six most promising advanced reactor systems, their associated fuel cycle concepts and the research necessary to advance these concepts for potential commercialization. Among the promising reactor technologies considered by the GIF, the Lead-cooled Fast Reactor (LFR) was identified as a technology with great potential to meet needs for both remote sites and central power stations.

This document deals with the LFR safety assessment and is based on the work carried out by the GIF-LFR provisional System Steering Committee (pSSC), briefly described in the following.

The GIF-LFR-pSSC was initially formed in 2005. The original membership included the EC, the US, Japan and Korea. With Korea primarily in observer status between 2005 and 2008, this initial committee worked together to prepare a series of drafts of an initial LFR System Research Plan (LFR-SRP), among its other activities.

In 2010, an MOU was signed between EC and Japan, and this resulted in a reformulation of the pSSC. Then in 2011, the Russian Federation added its signature to the MOU. In April 2012, the reformulated pSSC met in Pisa, Italy and a number of actions were defined. The US was invited to participate in the activities of the pSSC in observer status, and the process of preparing a revised SRP was initiated. The new pSSC, with representatives of EC, Japan and Russia, envisioned various updates to the SRP and actively participated in the development of the revised GIF Technology Roadmap (issued in 2013), and (also in 2013) both China and Korea joined the pSSC in observer status.

The reference concepts that form the basis of the revised SRP are the following:

- The European LFR (ELFR) as a large, central station plant (600 MWe);
- The BREST-OD-300 (300 MWe) as a medium size plant;
- The Small Secure Transportable Autonomous Reactor (SSTAR) as a small plant.

As it can be easily understood, the development of this White Paper requires a specific design reference since most of the analyses are design specific. During the March 2013 pSSC meeting in Paris the committee decided to use as a reference for this work the ALFRED (Advanced Lead Fast Reactor European Demonstrator) design, a scaled down reactor with respect to the industrial central station plant, the ELFR. Both designs, ELFR and ALFRED, have been carried out as part of the activities of the 7th Framework Program (FP7) LEADER (Lead-cooled European Advanced Demonstration Reactor) Project.

The decision to use ALFRED as the example LFR as the basis for the White Paper development was motivated also by the fact that, for this system, a consistent set of information has been disclosed and is available for the application (thanks to all the partners of the LEADER project).

It is important to point out however that general safety considerations and many design aspects of the three reference systems for the LFR-pSSC activities are common across the GIF - LFR systems and implemented also in ALFRED. As a consequence qualitative considerations should be perceived as generally shared between different LFR systems while the ALFRED reference will be used in the following when the application of the methodology becomes design specific for quantitative considerations.

In the following, brief descriptions of the main characteristics of GIF-LFR reference systems are presented followed by a more detailed description of the ALFRED design.

2.1 The European Lead-cooled Fast Reactor ELFR

Beside obvious differences related to size, the three systems taken as references for the GIF-LFR-PSSC activities share a significant number of technical issues and many common features, especially as far as safety design is concerned; thus, there are many commonalities from the point of view of design and engineering of these systems and the solutions adopted.

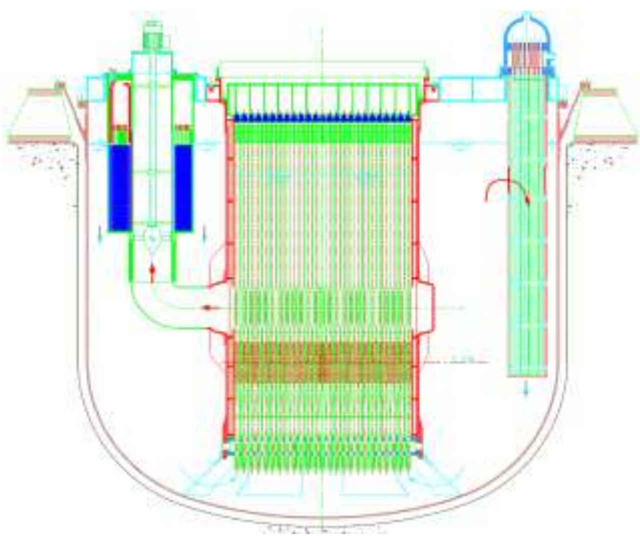

Figure 2.1: ELFR: the European Lead-cooled Fast Reactor

ELFR system is an evolutionary design representing a modification to the earlier ELSY reactor concept. Figure 2.1 provides an overview sketch of the ELFR reactor vessel and its contents. Several characteristics of the ELFR design are summarized in Table 2.1.

The ELFR primary system has a pool-type configuration, with the main vessels supported by a Y-support holding the main vessel in the upper part.

Table 2.1: ELFR Summary Parameters	
Power 1500 MW (th)	600 MW(e)
Core diameter	4.5 m
Core height	1.4 m
Core fuel	MOX (1st load)
Coolant temp.	400/480°C
Maximum clad temp.	550°C
Net efficiency	~42%
Core breeding ratio -CBR	~ 1

The Reactor Vessel (RV) has been kept as compact as possible, in order to reduce the coolant inventory and the corresponding seismic loads, while being of sufficient size to accommodate the required number of components (i.e. 8 Steam Generators (SGs), 8 Primary Pumps (PPs), and 8 Decay Heat Dip Coolers (DCs)). The hot pool of the ELFR vessel is enclosed by an Inner Vessel (IV), connected to the PPs through suction pipes. Each PP is installed at the centre of its corresponding SG, which transfers the heat from primary lead coolant to water-steam in a superheated cycle. The free level of the hot pools inside each SG/PP unit is higher than the free level inside the Inner Vessel, the different heads depending on the pressure losses across component parts of the primary circuit. The design is based on a core pressure loss of 0.9 bar and a total primary pressure loss of 1.4 bar. The core inlet and outlet temperatures are 400°C and 480 °C, allowing for a sufficient margin in the cold plenum from the freezing point of the lead coolant, while reducing the potential for embrittlement (for structures wetted by cold lead) and corrosion (for structures in hot molten lead). The maximum speed of the primary coolant is 2 m/s (10 m/s at the tips of the pump impeller) to limit erosion.

The internal reactor component arrangement and design presents a simple flow path for the primary coolant. The locations of the heat source (within the core) and of the heat sinks (SGs) allow for efficient natural circulation of the coolant under emergency shutdown conditions. Two safety systems for decay heat removal have been considered as an integral part of the design from the beginning of the activities. They are characterized by passive operation, diversity and redundancy while, in addition, being completely independent from one another. The design of the core has been driven by the implementation of the so-called "adiabatic" [17] reactor concept. The adiabatic reactor concept concerns the operation of a reactor with an equilibrium fuel cycle, so that the fuel composition remains the same between two successive loadings, ensuring the full recycling of all the actinides, with either natural or depleted uranium as only top-up/input material and fission products as well as reprocessing and fabrication losses as outputs, as illustrated below in Figure 2.2.

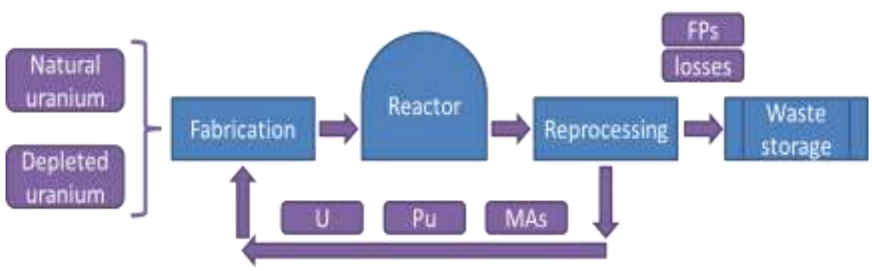

Figure 2.2: ELFR Adiabatic Fuel Cycle

This approach is conceptually very similar to that used for BREST-OD-300.

2.2 The BREST-OD-300 Lead-cooled Reactor

The BREST-OD-300 reactor is a pilot demonstration reactor (300 MWe) considered as a prototype of future commercial reactors of the BREST family for large-scale nuclear plants characterized by the idea of "natural safety." Figure 2.3 provides an overview sketch of the BREST-OD-300 system.

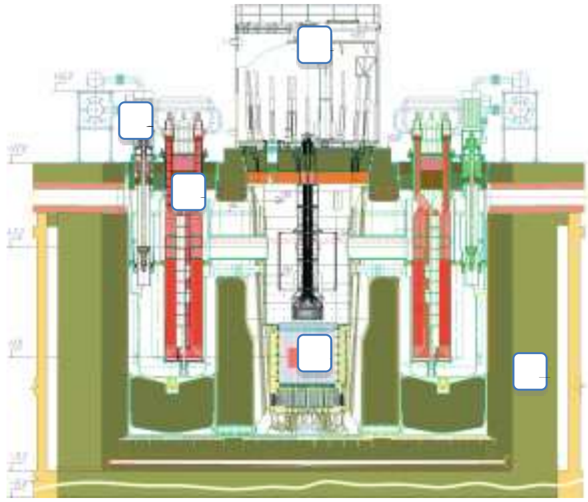

Figure 2.3: BREST-OD-300

Several of the relevant characteristics of the BREST-OD-300 design are summarized below in Table 2.2.

Table 2.2: BREST Summary Parameters	
Power 700 MW(th)	300 MW(e)
Core diameter	2.6 m
Core height	1.1 m
Core fuel	UN + PuN
Coolant temp.	420/540°C
Maximum clad temp.	650°C
Efficiency	43-44%
Core breeding ratio (CBR)	~ 1

BREST-OD-300 is a reactor facility of pool-type design, which incorporates within the pool the reactor core with reflectors and control rods; the lead coolant circulation circuit with steam generators and pumps; equipment for fuel reloading and management; and safety and auxiliary systems. The reactor equipment is arranged in a steel-lined, thermally insulated concrete vault.

BREST has a widely-spaced fuel lattice with a large coolant flow area, resulting in low pressure losses, favouring the establishment of primary natural circulation for decay heat removal. It shares with other designs the absence of uranium blankets, replaced by lead reflector with the proper albedo improving power distribution, providing negative void and density coefficients, and ruling out the production of weapons-grade plutonium. The BREST decay heat removal systems are characterised by passive and time-unlimited residual heat removal directly from the lead circuit by natural circulation of air through air-cooled heat exchangers, with the heated air vented to the atmosphere.

The fuel type considered for the first core of the BREST fast reactor is nitride of depleted uranium mixed with plutonium and Minor Actinides (MA), whose composition corresponds to that of irradiated (spent) fuel from PWR's following reprocessing and subsequent cooling for ~ 20 years.

The characteristics of lead allow for the operation with such fuel at an equilibrium composition. This mode of operation is characterized by full reproduction of fissile nuclides in the core (Core Breeding Ratio (CBR)~1) with irradiated fuel reprocessing in the closed fuel cycle. Reprocessing is limited to the removal of fission products without separating Pu and minor actinides (MA) from the mix (U-Pu-MA). One of the notable characteristics of the BREST plant is that a reprocessing plant is co-located with the reactor, eliminating in principle any accident or problem due to fuel transportation.

2.3 The Small Secure Transportable Autonomous Reactor (SSTAR)

SSTAR is a small modular reactor (SMR) that can supply 20 MWe/45 MWth with a reactor system that can be transported in a shipping cask. Some notable features include reliance on natural circulation for both operational and shutdown heat removal; a very long core life without refueling; and an innovative supercritical CO2 (S-CO2) Brayton cycle power conversion system. Figure 2.4 provides an overview sketch of the SSTAR system.

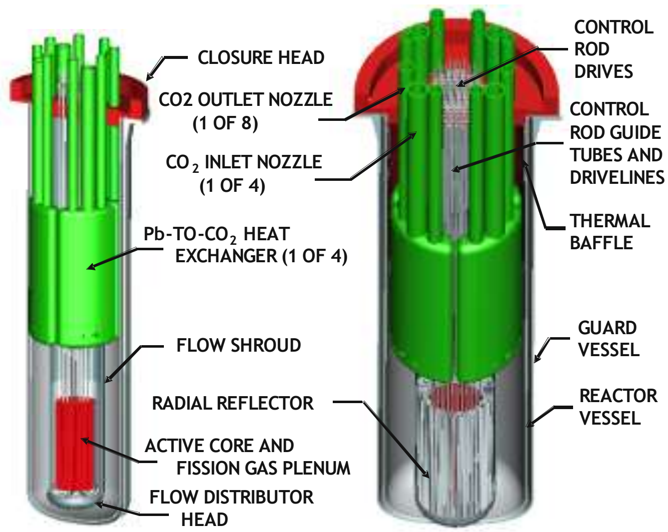

Figure 2.4: SSTAR

Several of the relevant characteristics of the SSTAR design are summarized below in Table 2.3.

The present pre-conceptual design of SSTAR is that of a small shippable reactor (12 m x 3.2 m vessel), with a 15-30 year life open-lattice cassette core and large-diameter (2.5 cm) fuel pins held by spacer grids welded to control rod guide tubes.

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.

Table 2.3: SSTAR Summary Parameters

Power 45 MW(th)	20 MW(e)
Core lifetime	15-30 years
Core fuel	Nitride - N_{15} enriched
Coolant temp.	420/567^{o}C
Maximum clad temp.	650^{o}C
Efficiency	44%
Core breeding ratio (CBR)	~ 1

Design features of the reference SSTAR in addition to the lead coolant, 15-30-year cassette core and natural circulation cooling, include autonomous load following without control rod motion, and use of a supercritical CO2 (S-CO2) Brayton cycle energy conversion system.

The incorporation of inherent thermo-structural feedbacks imparts a high degree of passive safety, while the long-life cartridge core life imparts strong proliferation resistance.

2.4 ALFRED design

During March 2013 meeting in Paris the LFR-pSSC decided to use as a reference for this work the ALFRED (Advanced Lead Fast Reactor European Demonstrator) design.

ALFRED is a scaled down reactor compared to the industrial prototype ELFR proposed in LEADER, which is on its turn based on the 600 MWe ELSY reactor. ALFRED has a relatively low power (125 MWe) with a compact design to reduce the cost but maintaining its representativeness. It is cooled by pure lead. For investment protection the design is based as much as possible on simple and removable components and operates at the lowest temperatures compatible with the pure lead coolant.

Primary system

The configuration of the primary system is pool-type. This concept permits containment of all the primary coolant within the Reactor Vessel RV), thus eliminating problems related to out-of vessel circulation of the primary coolant (Figure 2.5).

The Reactor assembly presents a simple flow path of the primary coolant, with a Riser and a Downcomer. The heat source (the Core), located below the Riser, and the heat sink (the Steam Generators - SG) at the top of the Downcomer, allow for efficient natural circulation of the coolant. The primary coolant moves upward through the pump impeller to the vertical shaft, then enters the SG through the lead inlet holes, flows downwards on the shell and exits the SG.

The free level of the hot pools inside the Steam Generator and Primary Pump units is higher than the free level inside the Inner Vessel (IV), the different heads depending on the pressure losses across component parts of the primary circuit. The volume between the primary coolant free levels and the reactor roof is filled by a cover gas plenum.

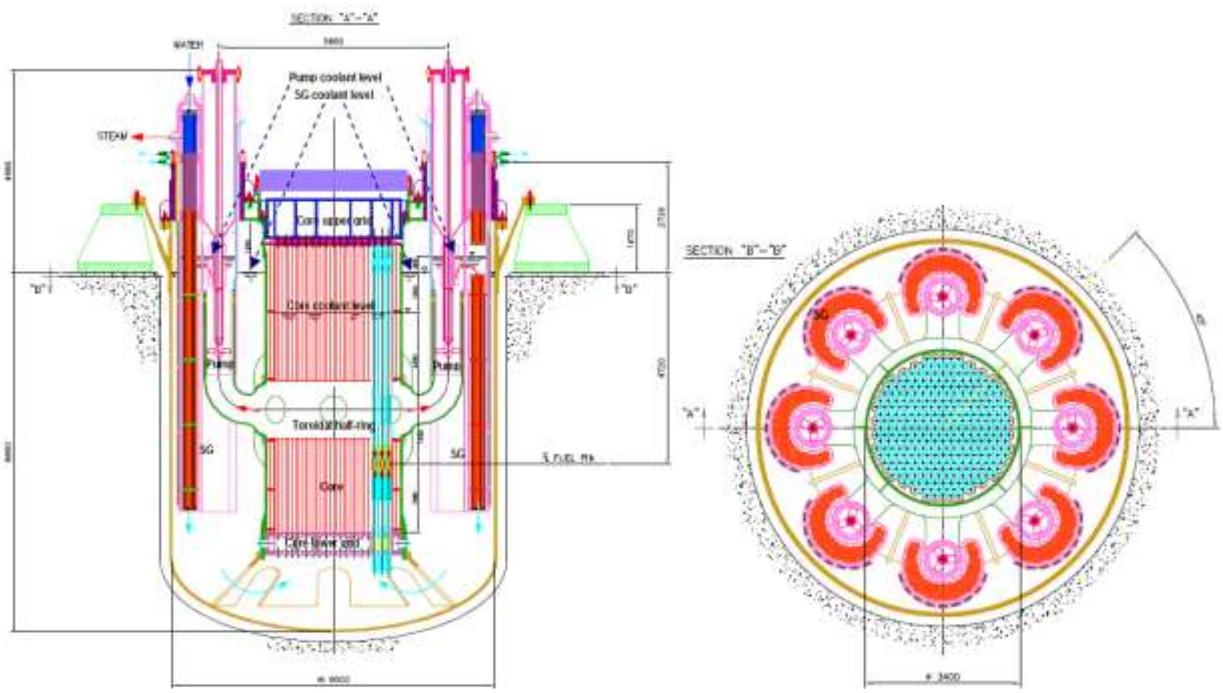

Figure 2.5: Lay-out of the ALFRED primary system: vertical and horizontal sections

Primary system arrangement

The **Reactor Vessel** is cylindrical with a torospherical bottom head. It is anchored to the reactor cavity from the top, by means of a vessel support. The upper part is divided in two branches by a "Y" junction: the conical skirt that supports the whole weight and the cylindrical one that supports the Reactor Cover. A cone frustum, welded to the bottom head, provides the function of bottom radial restraint of Inner Vessel.

A steel layer covering the reactor pit constitutes the **Safety Vessel** (SV). The dimensions of the gap between the safety vessel and the reactor vessel are sufficient for the In-service Inspection tools. The SV is cooled by the same system that cools the concrete of the cavity walls. This system is inserted inside the concrete and is independent from the reactor cooling systems. This design solution mitigates the consequences of through-wall cracks with leakage of lead: any reactor vessel leakage is discharged into the Safety Vessel. The RV and the SV are arranged in such a manner that, in case of a reactor vessel leak, the resulting primary coolant always covers the SG inlet and the lead flow path is indefinitely maintained.

The **Inner Vessel** has two main functions: Fuel Assemblies (FAs) support and separation between hot plenum and cold plenum. It is fixed to the cover by bolts and is radially restrained at the bottom. Lead flow is guided from the FAs outlet towards the PP inlet pipes by a toroidal half-ring. Moreover, the pipes that connect the hot zone with the inlet of the PP are integrated in the Inner Vessel. The cylindrical IV has a double wall shell: the outer thick wall has a structural function, while the inner thin wall follows the core section profile. The Core Lower grid is a box structure with two horizontal perforated plates connected by vertical plates. The plate holes are the housing of the FAs foots and the plate distances must be sufficient to assure the verticality of FAs. The diagrid is mechanically connected to the IV with pins (possible removal/replacement during reactor lifetime). The Core Upper grid is a box structure like the lower grid but stiffer. It has the function to push down the FAs during the reactor operation. A series of preloaded disk springs press each FA on its lower housing. A hole is present for each disk to allow the passage of instrumentation (i.e., thermocouples, neutron detectors, etc.).

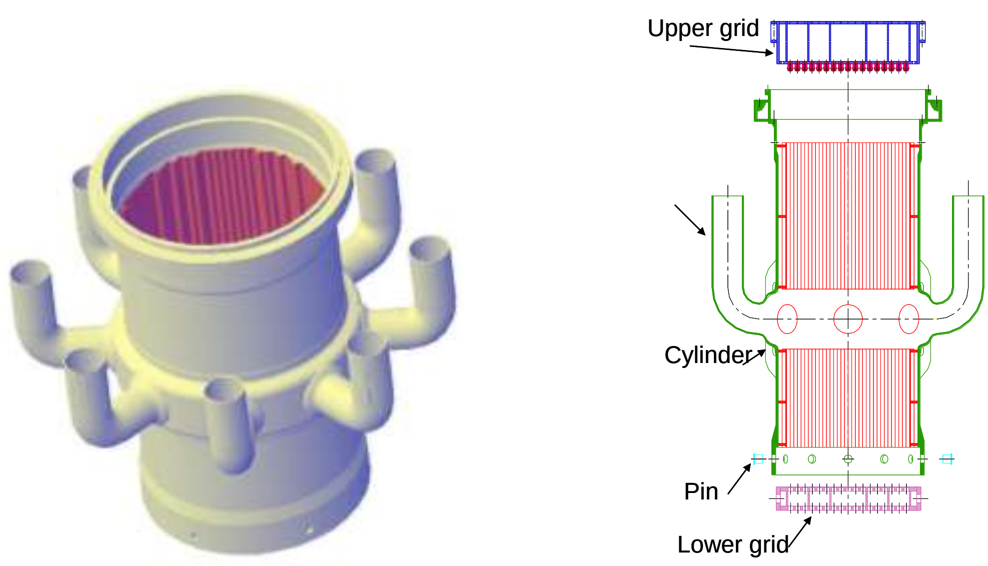

Figure 2.6: Inner Vessel: 3-D sketch and axial section with FAs inside

Core

The adopted core configuration is constituted by wrapped Hexagonal Fuel Assemblies. It utilizes MOX as fuel and uses hollow pellets and a low active height in order to improve natural circulation. The total power is 300 MWth. The core scheme is made of 171 FAs, 12 CR (Control Rods) and 4 SR (Safety Rods), surrounded by 108 Dummy Elements (ZrO_2-Y_2O_3) shielding the Inner Vessel. Each Fuel Assembly is about 8 m long and consists of 127 fuel pins, fixed to the bottom of the wrapper and restrained sideways by grids. Tungsten deadweight (Ballast) prevents buoyancy forces in lead. Upper elastic elements (cup springs) prevent lifting induced by hydrodynamic loads and accommodate axial thermal expansions. The FAs upper end extends beyond the lead free surface in the cover gas for easy inspection and handling. In this way it is possible to achieve refueling without the need of in-vessel refueling machines.

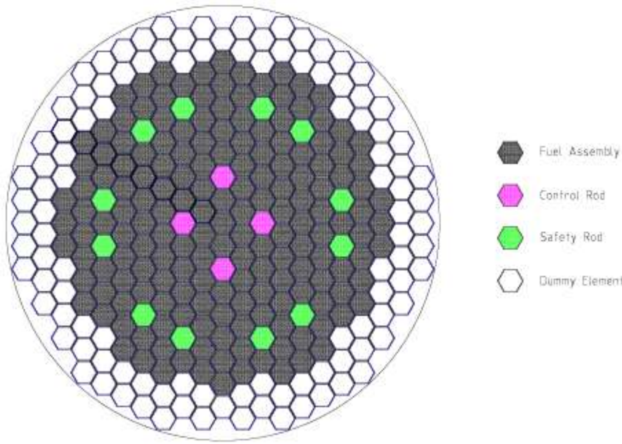

Figure 2.7 ALFRED Core

ALFRED is equipped with two diverse, redundant and separate shutdown systems (adapted from that under investigation in the frame of the CDT-MYRRHA project):

(1) The CR (Control Rod) system, used for both normal control of the reactor (start-up, reactivity control during the fuel cycle, power transients and shutdown) and for SCRAM in case of emergency. The Control rods are extracted downward and rise up by buoyancy in case of SCRAM. The control mechanism pushes the assembly down with a ball screw, placed, with its motor and resolver atop the cover (at cold temperature (<70°C)), and protected from radiation by a shielding block. The actuator is coupled to a long rod by an electromagnet. When the coupling electromagnet is switched off (in case of SCRAM), the absorber assembly and the rod are free to rise up. CRs use a 19 pins absorber bundle, cooled by the primary coolant flow. These pins are fitted with a gas plenum collecting the helium and tritium, produced by nuclear reaction of ^{10}B.

(2) The SR (Safety Rod) system, is the redundant and diversified complement to the control rods for SCRAM only. The absorber bundle stays in the primary coolant. The rod is extracted upward and is inserted downward against the buoyancy force. The absorber gets inserted by the actuation of a pneumatic system. In case of loss of this system, a tungsten ballast will force the absorber down by gravity in a slow insertion.

For both systems the materials considered are B4C enriched in ^{10}B at 90% as the absorber, T91 for the guide tubes, 15-15 Ti for the clad and ZrO2 (95%) - Y2O3 (5%) for the insulator and the reflector components.

Steam Generator and Primary Pump units

The steam generator and primary pump are integrated into a single vertical unit. Eight SG/PP units are located in the annular space between the cylindrical inner vessel and the reactor vessel wall. The primary pump is placed on the hot side of the steam generator, having its mechanical suction in the hot pool inside the inner vessel. The primary coolant moves upward through the pump impeller to the vertical shaft, then enters the SG through the lead inlet holes, flows downwards on the shell and exits the steam generator. The pump motor is located above the reactor roof.

Each SG consists of a bundle of 542 bayonet tubes immersed in the lead vessel pool for 6 m of their length. The bayonet tube is a vertical tube with external safety tube and internal insulating layer, composed of 4 concentric tubes (Figure. 2.8): a slave tube, an inner tube, an outer tube and an outermost tube. The internal insulating layer (delimited by the slave tube) has been introduced to ensure the production of superheated dry steam: in fact, without an insulating system, the high ΔT ($\approx 115°C$) between the rising steam and the descending feedwater would promote steam condensation in the upper part of the steam generator. The gap between the outermost and the outer bayonet tube is filled with pressurized helium and high thermal conductivity particles (such as synthetic diamonds) to enhance the heat exchange capability. In case of an external tube break this arrangement guarantees that the primary lead does not interact with the secondary water. Moreover, a tube break can be easily detected by monitoring the Helium gap pressure.

The Primary Pump is surrounded by the SG tube bundle, and its housing is the SG casing. The Pump is fixed to the top of SG casing by a bolted joint. This allows for easy removal of the component.

Primary Pump studies are in progress. Based on analyses performed during previous LFR projects (EUROTRANS, ELSY) an axial pump has been adopted. As far as the PP impeller material is concerned, Maxthal ceramic material has been proposed, but its reliability must be still demonstrated. An alternative solution can be a SS impeller with a ceramic coating.

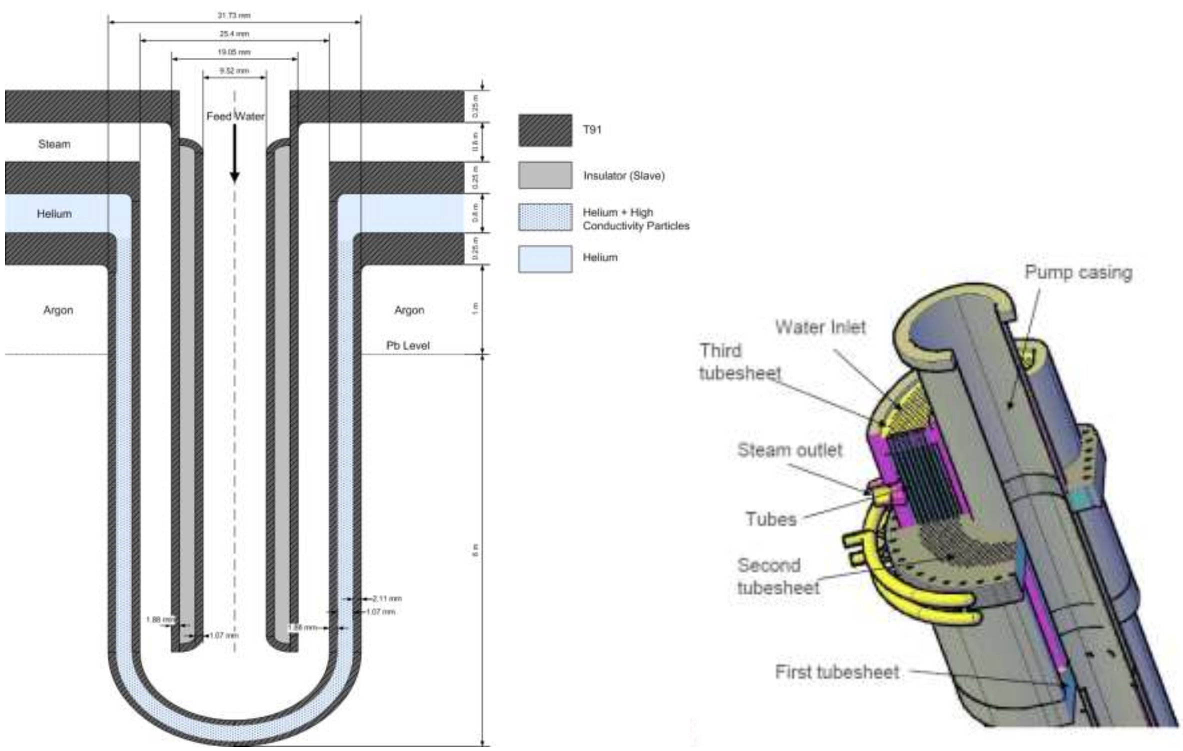

Figure 2.8: Bayonet tube configuration and SG 3D scheme

Decay Heat Removal System

The Decay Heat Removal system (DHR) consists of two passive, redundant and independent systems, DHR1 and DHR 2, both composed of four Isolation Condenser systems (ICs) connected to four SGs on the secondary side (i.e., one IC for each SG). The system design considers the single failure criterion, since three out of four ICs are sufficient to remove the decay heat power.

The DHRs are dedicated safety systems, not used for normal operation. The separation is achieved through placing the two DHRs in physically different locations. A physical structural barrier or another means of protection will be placed between adjacent ICs to ensure that failure of one of them could not harm another one. The diversity requirement has been relaxed (in any case, two DHR systems will be fabricated by diverse manufacturers) due to the high redundancy and considering that the SG tube bayonet concept allows a continuous monitoring of the SG status.

Both systems are completely passive, with an active actuation through valves equipped with redundant and diverse energy sources (batteries or locally stored energy). Each DHR system must be ready to operate after the reactor trip in order to remove the decay heat power, in case of unavailability of the normal path (i.e., the by-pass to the Condenser). The actuation logic guarantees the actuation of DHR1 first, whereas the DHR2 actuates only in the case of failure of the first system. Moreover, the total number of isolation condensers called to operate can never exceed the four units, in order to avoid an excessive cooling of the primary coolant leading to fluid solidification.

Each of the four independent IC sub-systems (Figure 2.9) consists of:

- One heat exchanger (isolation condenser), constituted by a vertical tube bundle with an upper and a lower horizontal header.
- One water pool, where the isolation condenser is immersed; the amount of pool water is sufficient to guarantee 3 days of decay heat removal operation.
- One condensate isolation valve (to meet the single failure criterion, this function must be performed at least by two parallel valves).

Each IC is connected to a SG: the upper header of the IC is connected to the main steam line and the lower header of the IC is connected to the main feed water line.

In normal operation, the isolation valve below the condenser is closed, the condenser is full of water and no heat exchange takes place. As the IC subsystem is called to operate, the feed water line and steam line are isolated and the condensate isolation valve opens. The subcooled water stored into the IC tube bundle drains into the SG, due to the hydrostatic head existing between the reactor and the IC. The IC tubes and headers become empty and their internal cold surface starts to condensate the steam coming from the SG and hence to transfer heat to the cold pool water. The steam condensation causes a pressure reduction which calls other steam from the SG. The water injected into the secondary system from the IC during the draining phase, vaporizes into the SG tube bundle contributing to the secondary system pressure rise: the safety relief valves continue to operate to reject to the atmosphere the excess of steam and to guarantee a secondary side pressure of 195 bar(a). When the IC reaches its steady state condition and starts to remove the thermal power from the primary coolant, the secondary side pressure rise decreases and finally stops leading to the closure of the safety relief valves on the main steam line.

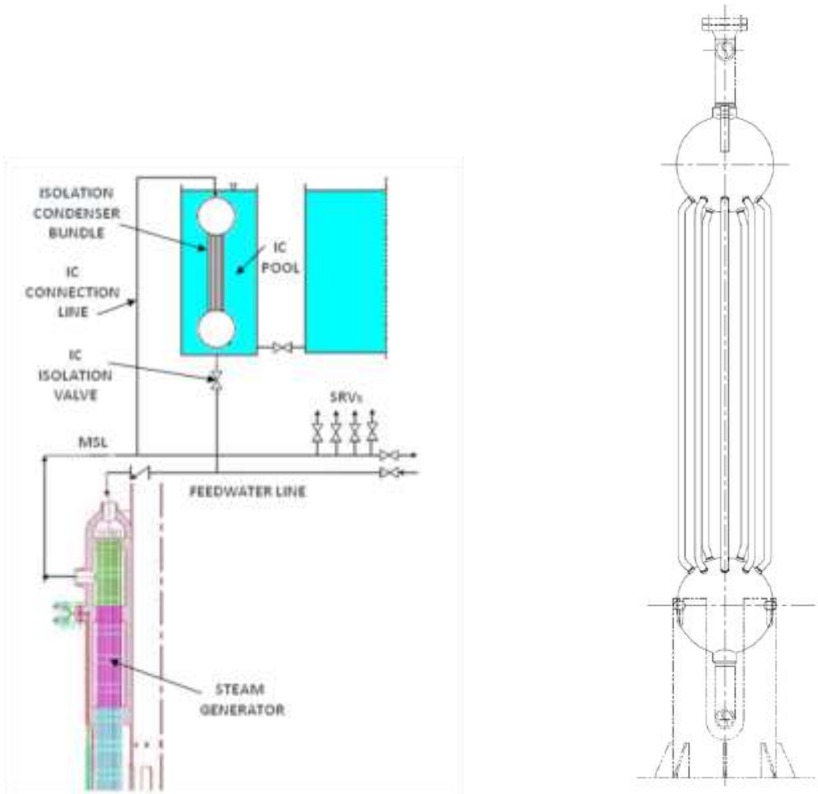

Figure 2.9: ALFRED Isolation condenser scheme and IC bundle

Secondary System

The Secondary System proposed for ALFRED is based on a dual turbine configuration with three extractions in the HP turbine and three more in the LP turbine, with an axial outlet. There will be reheating with steam from the first extraction and six preheaters supplied with steam from each turbine extraction, as well as a final heater supplied with main steam. This main steam shall be adequately throttled so that the feedwater temperature at the inlet of the steam generator (FWTC heater - feedwater temperature control) can be controlled. In addition, the de-aerator can be fed from the outlet of the HP turbine. The typical redundancy for the condensate and feedwater pumps (2x100% pumps) has also been considered.

An auxiliary lead heating system is added. This system would work when the power cycle is not in operation, in order to ensure the minimum temperature of the lead by transmitting heat from the secondary system if it is needed.

A 100% turbine bypass system is included, permitting direct transfer from the reactor to the condenser (bypass mode).

2.5 Summary of Gen-IV LFR reference systems and ALFRED parameters

The Lead-cooled Fast Reactor (LFR) features a fast neutron spectrum, high temperature operation, and cooling by a molten, low-pressure, chemically inert liquid metal with good thermodynamic properties. It would have multiple applications including production of electricity, hydrogen and process heat.

The LFR has excellent materials management capabilities since it operates in the fast-neutron spectrum and uses a closed fuel cycle for efficient conversion of fertile uranium. It can also be used as a burner to consume actinides from spent LWR fuel and as a burner/breeder with thorium matrices. An important feature of the LFR is the enhanced safety that results from the choice of molten lead as a chemically inert and low-pressure coolant. In terms of sustainability, lead is abundant and hence available, even in case of deployment of a large number of reactors. More importantly, as with other fast systems, fuel sustainability is greatly enhanced by the conversion capabilities of the LFR fuel cycle. Because they incorporate a liquid coolant with a very high margin to boiling and benign interaction with air or water, LFR concepts offer substantial potential in terms of safety, design simplification, proliferation resistance and the resulting economic performance. An important factor is the potential for benign end state to severe accidents.

For completeness in the following the performances of the three systems envisaged by the LFR-PSSC are listed with the expected performances of ALFRED.

The widely applied fundamental safety objectives and the Defence in Depth approach, as described by IAEA Safety Guides, have been preserved, since they have been in the Generation II and III nuclear power plants, characterized by a level of safety which is already very good. Moreover, although The European Utility Requirements (EUR) have been developed for LWR plants, the general requirements regarding the safety approach and the quantitative safety objectives have been adopted, in order to have a “minimum” target (EUR quantitative objective are more stringent than those of current plants) to be pursued in the development of the LFR plants.

Table 2.4: Summary Parameters of ALFRED vs. LFR-PSSC systems

	ALFRED	ELFR	BREST	SSTAR
Core power (MWt)	300	1500	700	45
Electrical Power (MWe)	125	600	300	20
Primary System type	Pool	Pool	Pool	Pool
Core Inlet T (°C)	400	400	420	420
Core outlet T (°C)	480	480	540	567
Secondary cycle	Superheated Steam	Superheated Steam	Superheated Steam	CO_2
Net Efficiency (%)	42	42	42	44
Turbine Inlet Pressure (bar)	180	180	180	20
Feed Temperature (ºC)	335	335	340	402
Turbine Inlet T (ºC)	450	450	505	553

Key safety objectives and safety principles adopted for the design

Further improvement of DiD, as suggested by INPRO and RWSG, has been taken into account in the LFR plant designs. In particular:

- Improving accident prevention, in particular by optimizing the balance between the measures taken at different levels of DiD and by increasing their independence:
    - **More robust design regarding system and component failures**: LFR design increases robustness against certain potential hazards through the avoidance of loop breaks (cooling loops inside the vessel), low primary side pressure (use of molten lead with high boiling point and low vapour pressure), extensive use of passive systems with potentially higher reliability;
    - **Capability to inspect**: LFR design simplification may be considered an essential feature in terms of in-service inspection capability taking into account the difficulties introduced by the high melting point of lead.

  - grace period until human actions: the LFR pool configuration and high thermal capacity naturally provide an extended grace time before operator action.

- Increased thermal inertia;
- Optimized human-machine interfaces;
- Extended use of information technology;
- Reduced complexity;
- Improved maintainability;
- Expanded use of passive features;
- More systematic consideration of the possibilities of multiple failures in the original plant design.

2.6 Research challenges linked to a lead-fast reactor

References [5] and [6] have provided the input of the present section and offer further reading and references to the issue.

The choice of lead as a coolant is motivated by its high boiling point of 1 749°C that makes coolant boiling during accidental conditions very unlikely, further by its high thermal inertia and its good neutronic as well as natural convection characteristics.

Additionally, lead is characterized by low, non-exothermic chemical activity in contact with air and water and this provides an opportunity for the elimination of the intermediate circuit, reducing number of components and hence possibly also costs. Consequently, in the ALFRED demonstrator (as in most of the LFR designs), SGs are located directly in the primary reactor vessel.

Therefore, water interactions with lead coolant, in case of steam generator tube rupture, need to be carefully considered and potential consequences analyzed. The gradual pressurization of the vessel and steam transport towards the reactor core with the potential for a reactivity insertion is the most credible threat to be met by design provisions.

On the other hand, sloshing and steam explosion are expected not to pose credible concerns as indicated by experimental evidence [6]. In any case, further experimental and analytical assessments are necessary to obtain data and improve understanding of effects and potential consequences of SG tube ruptures.

There is no operating experience and feedback on reactors cooled by pure lead. About 80 reactor years of experience and feedback were accumulated during operation of lead-bismuth eutectic cooled reactors used for Alfa/Lira-class submarines and land-based facilities in the former Soviet Union. The related feedback as well as experience from licensing of these reactors is, however, not easily available.

On the basis of analyses performed in SARGEN IV Deliverable D2.2 [5], which took into account conclusions obtained in the ELSY and LEADER Euratom Framework Programme Projects, specifically also for the ALFRED technological demonstrator, the main safety features and issues for LFRs can be summarized as follows:

Materials and coolant

- Molten lead is corrosive and is oxidized if oxygen concentration is not controlled;
- Molten lead might erode structural materials, metallic impurities produced by corrosion can be transported in the primary system;

- Lead vapors are chemically toxic;
- Large specific weight of lead and total primary inventory might, in case of external excitations, challenge structural integrity of systems or components;
- Large quantities of coolant in the main vessel of pool LFRs may lead to complex flow patterns and interactions between the coolant and structures;
- Lead is optically opaque.

The specific aspects related to the main safety functions: reactivity control, heat removal and confinement of radioactivity, are:

Reactivity control

- Ruptures of steam generator tubes might lead to over-pressurization of the primary side, sloshing and steam/water entrainment resulting in a positive reactivity insertion;
- Lead density reactivity coefficient might be positive in some core regions;
- Loss of core geometry (core compaction) might lead to a positive reactivity insertion and power increase.

Heat removal safety aspects

- Lead has high freezing point (327°C) with a potential for coolant solidification. Mechanical stresses might be exerted on structures during unfreezing if a proper melting sequence is not applied;
- Accumulation of corrosion products might lead to coolant blockages;
- High pressure of water decay heat removal systems;
- Natural convection.

Confinement of radioactivity:

- Corrosive properties of molten lead could challenge confinement barriers, in particular the cladding of the fuel pins.

3. Overview of the Safety Architecture's characteristics and performances

Within the FP7 LEADER project the provisions for safety functions have been a key issue in the design of subsystems of the demonstrator ALFRED. In addition, a systematic approach towards demonstrating the safety properties has been made in one of the LEADER work packages that was directed towards the identification of representative DBC (Design Basis Condition) and DEC (Design Extension Conditions) accident initiators [7]. This report describes the application of the following techniques:

- The OPT method was applied regarding the three critical safety functions, covering DiD levels 1 to 4.
- Initiating events were identified through a Master Logic Diagram top-down approach, and complemented with a categorization of the IEs according to their probability.
- Finally the accident initiators were categorized according to their probability and corresponding accident transients were identified to guide the DPA analyses to be carried out within LEADER.

3.1 Qualitative Safety Features Review (QSR)

The following considerations about qualitative safety features are common to all reference systems of the GIF activities.

Lead is peculiar among the coolants available for nuclear reactor systems for a number of reasons. As a dense liquid (when maintained above its normal melting temperature), it has excellent cooling properties while its nuclear properties (i.e., its low tendency to absorb neutrons or to slow them down) enable it to readily sustain the high neutron energy spectrum needed in a fast reactor while providing the reactor designer with great flexibility. These characteristics enable improved resource utilization, longer core life, effective burning of minor actinides, and open fuel pin spacing, important features in achieving sustainability, proliferation resistance, fuel cycle economics and enhanced passive safety by reducing void reactivity and enabling fuel cooling by natural circulation.

As mentioned earlier, lead has a very high boiling temperature, namely 1749 °C. As a result, the problem of coolant boiling is for all practical purposes eliminated. The high margin to boiling as well as low or negative void reactivity leads to important safety advantages that should also result in design simplifications and improved economic performance.

As a coolant operating at atmospheric pressure, the loss of coolant accident (LOCA) can be virtually eliminated by use of an appropriately designed guard vessel. This is not only a safety advantage, but also offers additional potential for plant simplification and improved economic performance since the complex process of simultaneous management of temperature, pressure and coolant level (as is seen in water-cooled reactors) is not necessary.

One of the most important characteristics of lead as a coolant is its chemical inertness. In comparison with other coolants, especially sodium and water, lead presents a benign coolant material that does not support chemical interactions that can lead to energy release in the event of accident conditions. Further, the tendency of lead to retain fission products and other materials that might be released from fuel in the event of an accident is another important advantage. The elimination of the need for an intermediate coolant system to isolate the primary coolant from the water and steam of the energy conversion system represents a significant advantage and potential for plant simplification and improved economic performance.

Following the Fukushima-Daiichi reactor accidents, it is important to consider future reactor technologies in light of the severe conditions that may take place even if their probabilities are taken to be very low. The PSSC considers that the LFR can demonstrate superior features to avoid the consequences of such a severe set of accident scenarios. The Committee points out that one of the primary problems in the Fukushima scenario was the common mode failure of diesel generators (caused by the tsunami) during an extended blackout condition (caused by the earthquake). An LFR would not need to rely on such backup power and would be resilient in the face of blackout conditions by virtue of passively operated decay heat removal enabled by the natural circulation capabilities of the lead coolant and its inertness in contact with air or water.

Second, the loss of primary coolant at the Fukushima-Daiichi reactors resulted from pressurized water coolant. An LFR with guard vessel would not suffer a loss of primary coolant, even in the event of a failure of the reactor vessel.

Third, the steam-cladding interactions at the Fukushima-Daiichi reactors resulted in the liberation of hydrogen and associated explosions. With the chemical inertness of lead as a coolant, such hydrogen generation could not occur.

The demonstrator design elaborated in the LEADER project aims at making full use of the beneficial characteristics of the lead coolant and addresses many of the safety challenges described in Section 2.6. Principles like "selecting robust and simple design solutions" are followed. The focus of [8] is on reactor/core design and the design of primary cooling system components (steam generator, main coolant pump).

3.2 Phenomena Identification and Ranking Technique (PIRT)

No explicit and systematic PIRT has been fully developed at this time. Even so, much of the approach foreseen in PIRT is implicitly contained in the building of the OPT in the following section.

3.3 Objective Provision Tree (OPT)

The OPT method, as a top-down method with a tree structure, has been applied to the ALFRED concept, taking account of:

- the levels 1 to 4 of DiD and
- the fundamental safety functions (i) control of reactivity, (ii) removal of heat from the fuel, and (iii) confinement of radioactive materials.

Within the exercise:

- the possible challenges to the safety functions;
- the plausible mechanisms which can materialize these challenges;
- the provided provision(s) to prevent, control or mitigate the consequences of the challenges/mechanisms;

have been identified. The description of DiD levels below is taken from reference [7].

DiD level 1

The objective for Level 1 is the prevention of deviations from normal operation, the prevention of failures, and to ensure that the safety systems would operate reliably if called upon at higher levels of defence.

The essential means are the conservative design and high quality in construction and operation.

Provisions taken to ensure SF2 (heat removal from the core) are displayed in the respective OPT tree in Figure 3.1. As an example, a design provision helping to ensure proper core heat removal is the high quality and preventive maintenance and inspection of fuel elements and of the components inside the vessel: this minimizes the occurrence of mechanisms causing a flow blockage.

The OPT trees provide a good view of the large number of provisions to be taken to ensure DiD level 1.

DiD level 2

The objective for Level 2 of Defence-in-Depth is the control of abnormal operation and detection of failures.

The essential means are control, limiting and protection systems and other surveillance features. The successful performance of Level 2 provisions will bring the plant back to normal operating conditions as soon as possible.

Features of Level 2 should come into play whenever a significant deviation from normal operation conditions occurs, implying insufficient safety provisions at Level 1 and the occurrence of a Postulated Initiating Event.

ALFRED is equipped with an Instrumentation and Control System, with the function to:

- assure the adequate monitoring of the status of the main plant parameters (primary pumps velocity, coolant temperature, core flow rate, secondary side pressure, temperature and flow rate, coolant inventory)
- establish and maintain the plant operating conditions within prescribed limits. The control systems automatically regulate the operating conditions in response to changing plant conditions, such as:
  - Reactor power control system, achieved by varying the position of the control rods, to control the reactor power and the coolant average temperature
  - Feedwater control system, to regulate the flow rate and the temperature of water at the inlet of steam generators
  - Steam dump control system, the system is designed to divert the main steam to the condenser, bypassing the turbine, in order to prevent Steam generator overpressure and to provide an additional thermal load, during plant startup or shutdown.

An example for provisions at this DiD level are control systems for reactor power and feed water flow/temperature that allow to automatically regulate, within prescribed limits, the operating conditions in response to changing plant conditions. In this way, abnormalities in the safety function heat removal are addressed (Figure 3.2).

DiD level 3

The objective for Level 3 of defence-in-depth is the control of accidents within the design basis. The essential means of protection are: inherent and engineered safety features, and accident procedures.

Despite of the provisions for prevention and control of abnormal occurrences (failure of Levels 1 and 2), accident conditions may still occur. Inherent safety features and protection systems and, if needed, engineered safety features, are provided to prevent evolution toward severe plant conditions and to confine radioactive materials.

Design and operating procedures are aimed at maintaining the effectiveness of the barriers, in the event of such postulated accidents. Inherent features, as well as active or passive systems, are used.

In the short term, all these lines of defence are actuated inherently or by the reactor protection system when needed. The engineered systems (active and passive) are designed on the basis of design criteria to ensure a high reliability:

- redundancy (single failure criterion);
- prevention of common mode failure due to internal or external hazards, by physical or spatial separation and structural protection;
- prevention of common mode failure due to design, manufacturing, construction, commissioning, maintenance or other human intervention, by diversity or functional redundancy;

- automation to reduce vulnerability to human failure, at least in the initial phase of an incident or an accident;
- testability to provide clear evidence of their availability and performance;
- qualification for specific environmental conditions that may result from an accident or an external hazard.

An example of a provision on this DiD level is the decay heat removal during accidents by means of Isolation Condenser System connected to the steam generator. The system is completely passive, with an active actuation (through valves alignment) that is however equipped with redundant and diverse energy sources (batteries or locally stored energy).

In ALFRED, considering that the bayonet concept allows a continuous monitoring of the SG status, the diversity requirement is relaxed: the two redundant DHRs are both constituted by 4 Isolation Condensers (three out of four guarantee the decay heat removal) (Figure 3.3).

DiD level 4

The objective for Level 4 of defence-in-depth is the control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents. The essential means are complementary measures and accident management.

Since lead density is slightly higher than that of the oxide fuel, fuel dispersion dominates over fuel compaction, reducing considerably the likelihood of the occurrence of severe re-criticality events in the case of core disruption. In fact, the dispersion of fuel and molten clad from core active zone is enhanced resulting in a self-reducing extension of core degradation.

Hence, severe plant conditions for ALFRED do not necessarily involve large fuel melting but, in any case, conditions that could challenge the structural integrity of the core and thus the ability to analyze the course of an event are to be managed.

Consideration is given to severe plant conditions that were not explicitly addressed in the design (insufficient provisions at Levels 1 to 3) owing to their very low probabilities. Such plant conditions may be caused by multiple failures or by an extremely unlikely event such as a severe earthquake.

The large thermal inertia of the plant and characteristics of the fuel and reactor internal structures will provide considerable time to deal with these conditions. If necessary, additional measures and procedures may be provided. Ancillary and support systems, if employed, would be designed, manufactured, constructed and maintained consistent with the required reliability.

Measures for accident management are also aimed at controlling the course of severe plant conditions and mitigating their consequences.

Essential objectives of accident management are:

- to monitor the plant status;
- to maintain core sub-criticality;
- to protect the integrity of the RV by ensuring heat removal from the core and preventing excessive loading conditions (both thermo-mechanical and chemical);
- to limit the release of radioactive material to the environment;
- to regain and maintain control of the plant.

In the very unlikely event with loss of all heat sinks (both DHRs trains and secondary systems), the heat can be extracted injecting water in the reactor cavity between the reactor and safety vessels, while in case of reactor vessel breach the decay heat can still be removed by the same system that cools the concrete of cavity walls.

Such very ultimate provisions are possible because lead is chemically inert with air and water so that fire fighters or dedicated rescue troops can flood the reactor with water.

The corresponding OPT for SF2 and DID level 1-3 are displayed in Figures 3.1 to 3.3.

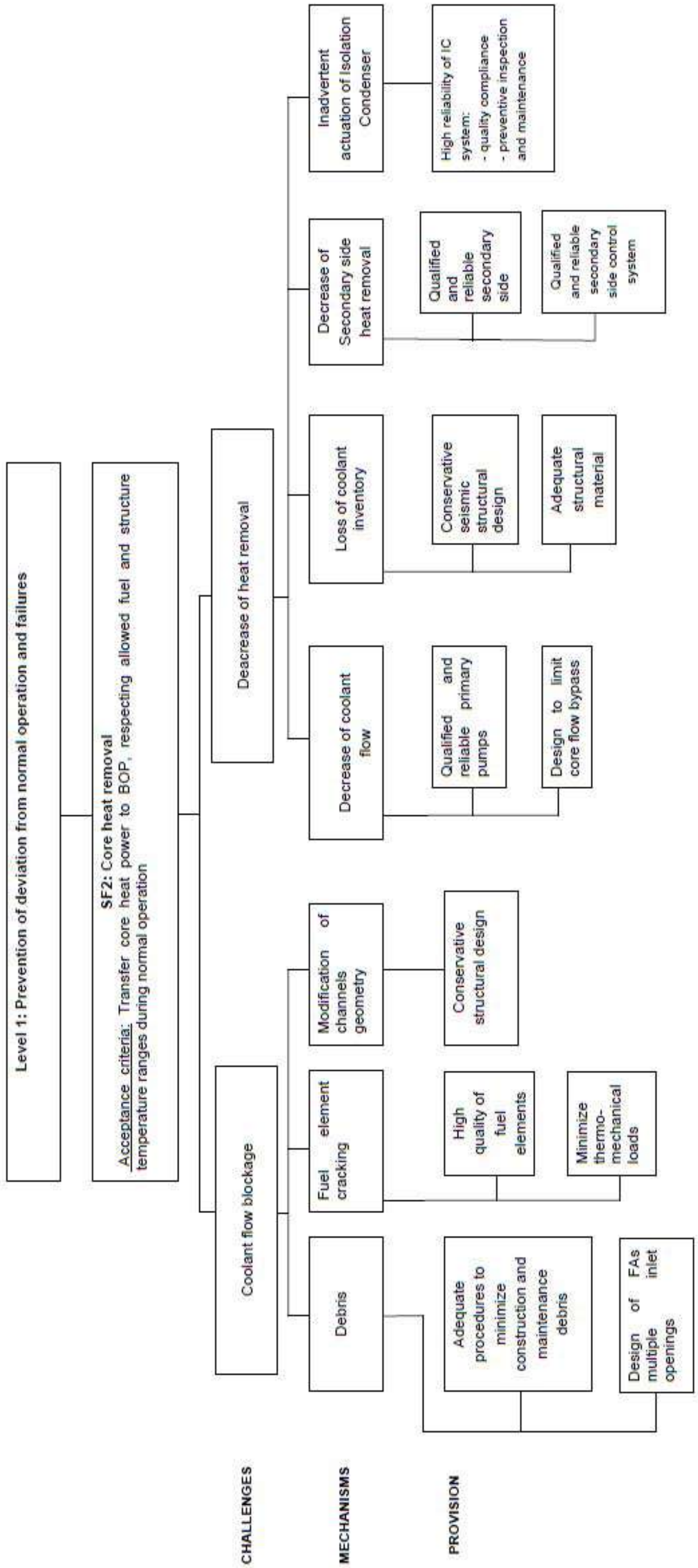

Figure 3.1: OPT-DiD Level 1 for SF2: Core Heat Removal

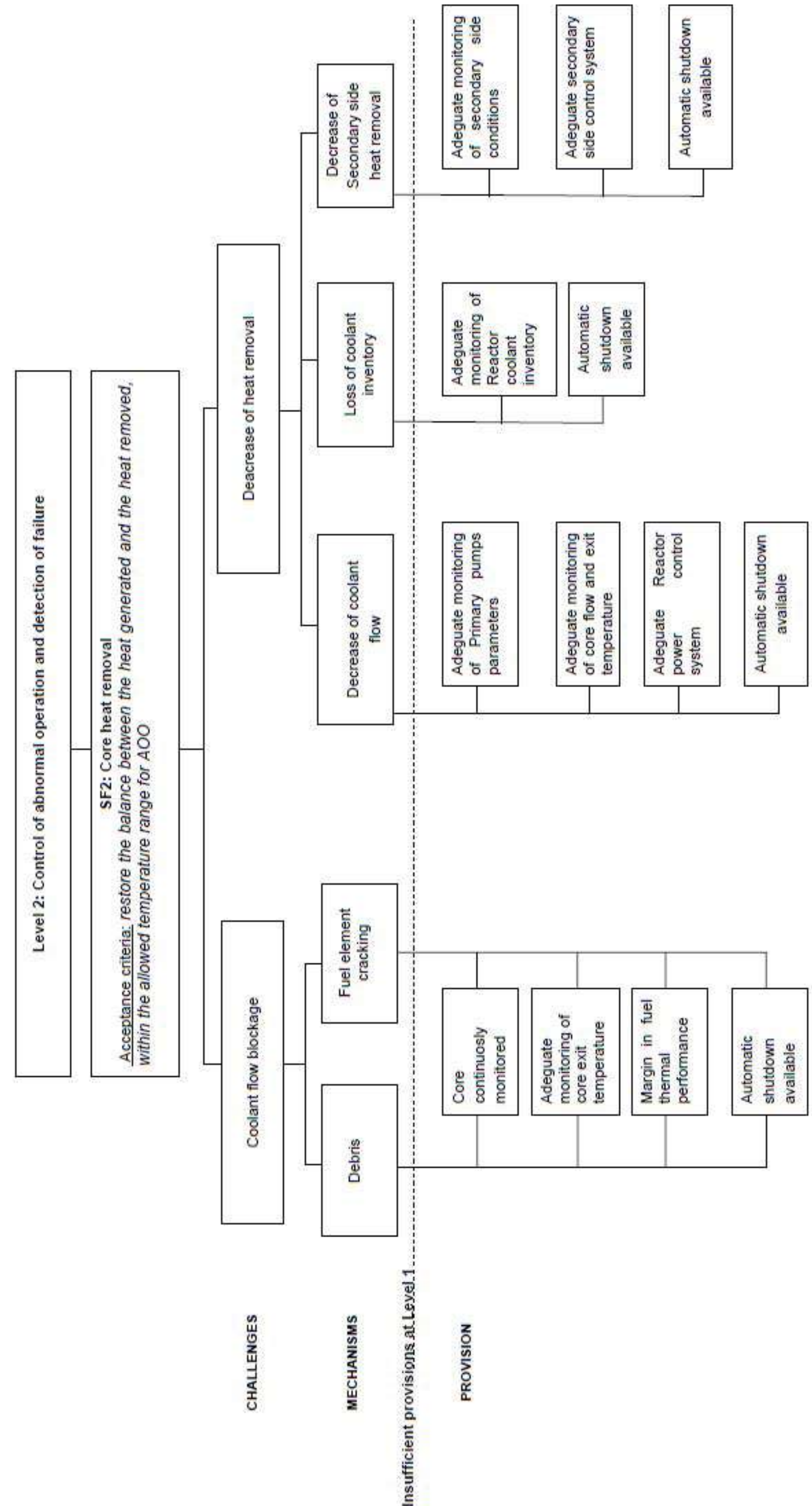

Figure 3.2: OPT-DiD Level 2 for SF2: Core Heat Removal

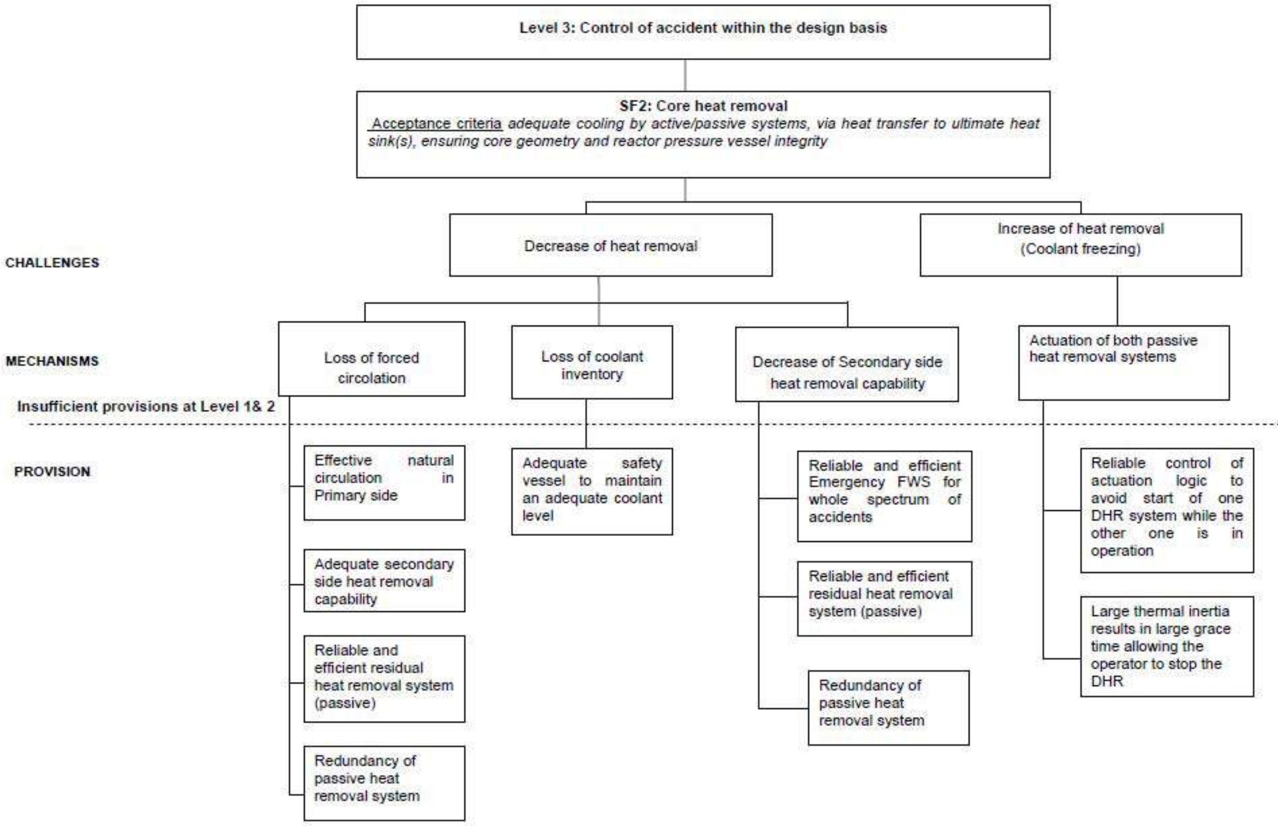

Figure 3.3: OPT-DiD Level 3 for SF2: Core Heat Removal

3.4 Deterministic and Phenomenological Analyses (DPA)

The extensive DPA use within LEADER project includes the safety analysis of the design basis condition (DBC) events [10] and design extension condition (DEC) events [11].

The postulated DBC events are those in which the reactor is assumed to shut-down normally upon demand, while DEC events are very unlikely events that foresee possible multiple failures or the failure of safety or mitigating systems, such as the reactor scram in the so-called unprotected transients (e.g. the reactor is assumed not to shut-down upon demand due to an assumed reactor control system failure).

Subtask 5.3 of the LEADER project was directed at identifying representative accident initiators for Design Basis Conditions (DBC) and Design Extension Conditions (DEC), including complex sequences, severe accidents and limiting events. By varying assumptions on the availability of different plant systems, a list of 25 potential accident transients was developed.

While this list is known to omit some transients, it was tried to avoid an exaggerate multiplication of the transients to simulate and the selected transients are estimated to cover most of the incidental and accidental situations to be taken into consideration for the safety analysis of ALFRED.

From the mentioned 25 transients a total of 19, considered the most challenging for the plant, were selected to be analyzed in the LEADER Tasks 5.4 and 5.5.

Examples of transients within different DBC are:

Transient	IE	Description	Category
TR-1	Drop (or insertion) of one control rod	Results in the perturbation of core radial power distribution. This is an asymmetric transient: it will be analyzed only if a suitable code will be available	DBC2
TD-7	Loss of all primary pumps	The reactor trip should be provided by pumps under-speed signal	DBC3
TRB-1	Steam system piping break	Break in the steam system piping at SG outlet (1 & 8 loops affected – sensitivity study) Reactor is tripped	DBC4
T-DEC1	Complete loss of forced flow + Reactor trip fails (total ULOF)	All primary pumps are stopped Feedwater system available Reactor trip fails	DEC

The results of the application of DPA on the representative protected DBC and DEC transients demonstrate that the Pb-cooled ETDR (ALFRED) plant is a very robust and forgiving reactor design if the feasibility of the proposed technological innovations can be demonstrated in the future by an extensive dedicated R&D program. This is ascribable to the combination of:

- the inherent, large thermal inertia of the Pb-cooled primary system,
- the detailed focus on the optimization of all safety relevant systems, in particular emphasizing appropriate design of all relevant control and safety systems (as well as components), and
- optimizing the neutronic core characteristics of the ALFRED “core system” thereby assuring various reactivity feedback effects (fuel, diagrid, pads, Pb-coolant and CRs drivelines expansion reactivity) that effectively depress the reactor power under all adverse DEC transient conditions,
- the ALFRED core design specifically set out to accommodate the ULOF transient.

3.5 Probabilistic Safety Assessment (PSA)

While the approach towards identifying transients to be analysed, in the section above, has introduced the probabilities of initiating events (IEs) into the safety approach, the PSA work on ALFRED has been limited to a description of the requirements for a probabilistic safety approach [9].

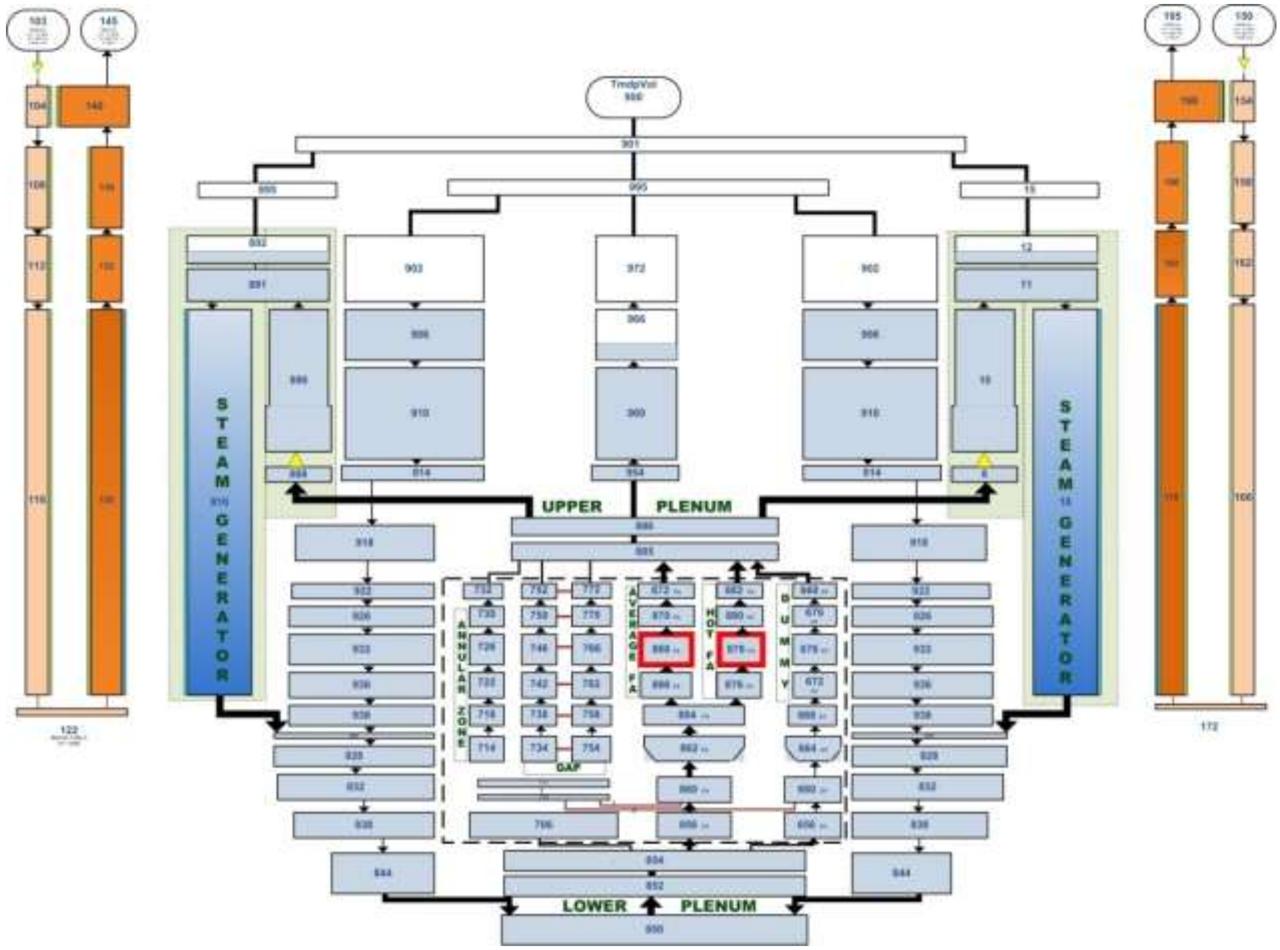

Figure 3.4: RELAP5 nodalization of ALFRED

4. Current System Development Status

The LEADER project under the EU's 7th Framework Programme was completed in April 2013. Besides developing an integrated strategy for the LFR development and improving the conceptual design of ELFR, an industrial size plant, LEADER was oriented towards proposing a design for the LFR demonstrator ALFRED.

For ALFRED, solutions were sought that are as close as possible to the adopted reference conceptual design, but meet the essential need to proceed to construction in a short time frame.

The future of ALFRED is strongly dependent on available funds. If the correct level of financing is provided the present plan, including the development phase of the technology and its application in ALFRED, estimates a reasonable date for ALFRED start of operation after 2025. The site has been already identified in Romania.

While ELFR and SSTAR can be considered at the present time as reference conceptual reactors, the most advanced and close to realization is the BREST-OD-300. Operation of the reactor is expected by 2021, and on that basis the LFR Technology Road Map has been consequently updated.

5. Conclusions: System Issues, Concerns and Benefits

Important safety issues of the lead technology have been noted in Section 2.6. Of these, the most significant seem those related to reactivity control and to corrosion. In addition, the high temperature and the opaqueness of the lead in the pool make handling complicated and require R&D for understanding and mastering system challenges.

This report is based on safety assessment work carried out within the FP7 LEADER project and the support of FP7 SARGEN IV project. The exercise of developing OPTs for the "degradation/disruption of the heat transfer path" that is challenging the core heat removal function has demonstrated that OPT is powerful in structuring safety issues and thus helping to define which safety analyses are to be carried out.

It has to be said that, at the present conceptual design stage, the toolset of ISAM has not been fully applied, and that it needs to be further demonstrated as the project continues.

Finally, it is noted that a regulatory approach for licensing GEN IV reactor will be needed and that this issue has to be supported to prepare, and shorten the time needed for the construction of LFR Demonstrators.

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