LFR safety approach and main ELFR safety analysis results

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Abstract. This paper summarizes the approach to safety for the LFR systems, developed on the basis of the recommendations of the Generation IV International Forum (GIF) Risk and Safety Working Group (RSWG) and taking into account the fundamental safety objectives and the Defence-in-Depth approach, as described by IAEA Safety Guides, as well as the Safety quantitative objectives reported in the European Utilities Requirements (EUR). LEADER project activities are focused on the resolution of the key issues as they emerged from the 6th FP ELSY project attempting to reach a new industrial size European Lead-cooled Fast Reactor (ELFR) configuration. Apart from the safety approach, the main results of the ELFR safety transient analysis, where the most important design basis condition (DBC) and design extension condition (DEC) transient initiators were re-analyzed using the system codes RELAP5 (ENEA), TRACE-FRED (PSI), SIM-LFR (KIT) and SIMMER (CIRTEN), are summarized.

1. INTRODUCTION

The LEADER (Lead-cooled European Advanced DEmonstration Reactor) project, funded by the European Commission in the frame of 7th framework program, aims to the development to a conceptual level of a Lead Fast Reactor Industrial size plant and of a scaled demonstrator of the LFR technology - ALFRED.

The project started from the results achieved in the previous ELSY (European Lead-cooled SYstem) project (6th FP), during which a pre-conceptual design of an industrial plant (600 MWe) was developed. Safety analysis of the re-designed ELFR was performed addressing all the “weak” points in the LFR design that were determined during the previous ELSY project. All the most important design basis condition (DBC) and design extension condition (DEC) transients were repeatedly reanalyzed for the re-designed ELFR configuration and conclusions from this activity are presented in this paper.

2. SAFETY APPROACH FOR LFR PLANT

As one of the six currently developed and analyzed Generation IV reactor systems – LFR, follows the general guidelines of the Generation IV safety concept recommendations. Among the goals for future nuclear energy systems, improved safety and higher reliability are recognized as an essential priority in the development and operation of nuclear power plants. A global safety approach for the LFR reference plant has been assessed and the safety analyses methodology has been developed [1].

The fundamental safety objectives and the Defence-in-Depth (DiD) approach, as described by IAEA Safety Guides, have been preserved. The ideal outcome will be a design that optimizes both capital costs and safety by applying defence in depth where it will have the desired effect, but not to "over-design" in a way that adds cost but provides little additional value in safety.

The recommendation of the Risk and Safety Working Group (RSWG[1]) has been taken into account, in particular:

- safety is to be **"built-in"** to the fundamental design rather than **"added on"**;
- full implementation of the Defence-in-Depth principles in a manner that is demonstrably exhaustive, progressive, tolerant, forgiving and well-balanced (e.g. rejection of "cliff edge effects" and availability of a sufficient grace period and the possibility of repair during accidental situations);
- "risk-informed" approach - deterministic approach complemented with a probabilistic one;
- adoption of an integrated methodology that can be used to evaluate and document the safety of Gen IV nuclear systems - Integrated Safety Assessment Methodology (ISAM). In particular the Objective Provision Tree (OPT) tool is the fundamental methodology used throughout the design process. The OPT is a top-down method which, for each level of DiD and for each safety objective/function, identifies the possible challenges to the safety functions, their related mechanisms, and the provisions needed to prevent, control or mitigate their consequences.

3. ELFR PLANT REFERENCE CONFIGURATION

The configuration of the ELFR primary system (Fig. 1) is pool-type [2]. This concept permits to contain all the primary coolant within the Reactor Vessel, thus eliminating all problems related to out-of vessel circulation of the primary coolant.

The Reactor Vessel (RV) is cylindrical with a torospherical bottom head. It is anchored to the reactor cavity from the top, by means of a vessel support. A steel layer covering the reactor cavity, constitutes the Safety Vessel (SV). The primary coolant always covers the SG inlet so to indefinitely maintain the lead flow path. The volume between the primary coolant free levels and the reactor roof is filled with an inert gas.

The core is made of 427 hexagonal and wrapped fuel assemblies (FAs) and 24 control/safety assemblies. Each FA is about 10 m long and consists of 169 fuel pins, fixed to the bottom of the wrapper and restrained sideways by grids. To maintain each fuel element in its position, a tungsten deadweight (Ballast) counterbalances the lead buoyancy during refueling.

The Inner Vessel (IV) the first structure around the core, has two main functions: 1) Fuel Assemblies support, and 2) Hot and cold plenum separation.

The LFR plant is equipped with two diverse, redundant and separate shutdown systems: 1) gravity driven system (only shutdown) passively inserted by a pneumatic system (by depressurization) from the top of the core. In case of failure of the pneumatic system, the safety rods are equipped with tungsten ballast that forces the absorber down by gravity with a lower velocity; 2) control/shutdown system inserted from below the active core zone using the strong lead buoyancy.

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[1] The RSWG was formed in the frame of Generation IV International Forum (GIF) to promote a homogeneous and effective approach to assure the safety of Generation IV nuclear energy systems

The eight steam generators and primary pump are integrated into separate vertical units. The primary pump is placed in the centre of the flat-spiral type 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 and then transversally (radially) through SG tubes on the shell side out of the steam generator to the downcomer through perforated double-wall casing.

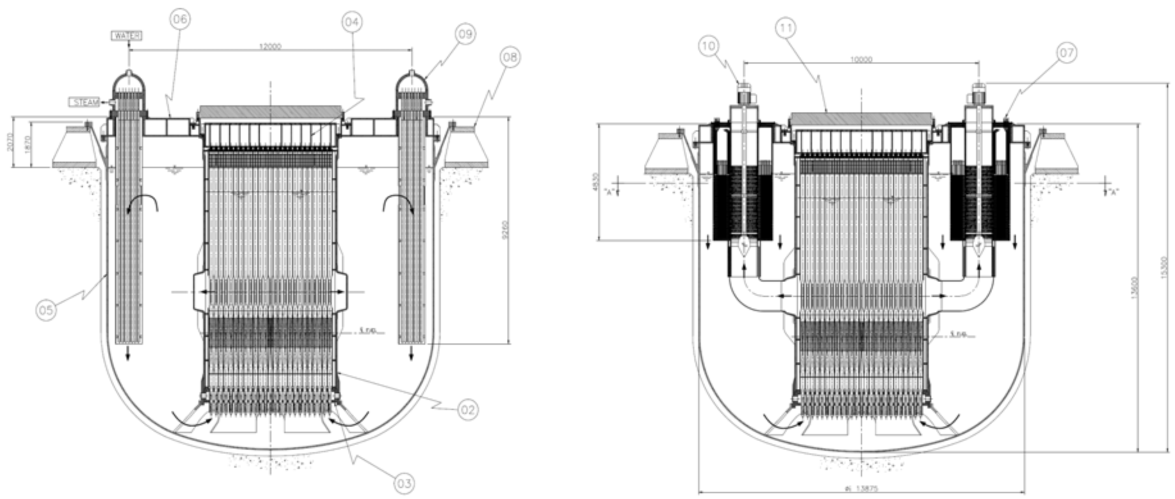


FIG. 1. Reactor block vertical sections: 01) Fuel assembly ;02) Inner vessel; 03) Core lower grid; 04) Core upper grid; 05) Reactor vessel ; 06) Reactor cover; 07) Steam Generator; 08) Vessel support; 09) DHR dip cooler; 10) Primary pump; 11) Reactor FAs cover)

The Decay Heat Removal system consist of two independent, redundant and diverse systems: 1) The Isolation Condenser System (IC) connected to the Steam Generator (i.e. four units provided on four out of eight plant steam generators); 2) DHR-2 System, constituted of four independent loops, equipped with eight dip coolers operating with water, immersed in the reactor pool. Both systems are completely passive, with an active actuation (valves). The IC system is the first line of defence whereas the DHR-2 comes into operation only in case of failure of the first one.

Each component inside the Reactor Vessel is removable and the fuel assemblies upper end extends beyond the lead free surface in the cover gas for refueling, without the need of in-vessel machines. This design solution is viable because the refuelling could be performed by opening the reactor cover (a flat steel plate with penetrations for the Steam Generator/Primary Pump units and Dip coolers) and accessing the fuel assemblies directly from the containment.

4. SUMMARY OF THE ELFR SAFETY ANALYSIS

As it was already mentioned before, safety analysis of the ELFR was performed testing all the "weak" points in the LFR design that were determined during the previous ELSY project. The most important design basis condition (DBC) and design extension condition (DEC) transients were repeatedly reanalyzed for the re-designed ELFR configuration.

The full list of the analyzed transients for ELFR is as follows:

- Protected loss of flow transient (PLOF),
- Unprotected loss of flow transients (ULOF),
- Unprotected loss of heat sink transient (ULOHS),
- Unprotected reactivity insertion transient (UTOP),
- Unprotected loos of flow and loss of heat sink transient (ULOF+ULOHS),

- Protected overcooling transient (OVC),
- Protected steam line break transient (SLB),
- Unprotected sub-assembly (SA) blockage transient, and
- Steam generator tube rupture (SGTR) accident.

The main results of all the analyzed unprotected transients are briefly presented in the following sub-sections of this paper.

4.1. Unprotected loss of flow transient (ULOF)

The Unprotected Loss of Flow (ULOF) accident is initiated by the drop of primary pump head with a halving time of the pump of 0.56 seconds. During the entire ULOF transient, the reactor protection system failed to operate, but the steam generator (SG) could work normally.

As the primary coolant flow starts to decrease (Fig. 2), the core power shortly increases to around 1600 MW, which is due to the positive reactivity feedback incurred by the coolant temperature increase. After the coolant temperature increase moves from core active region to core diagrids and pads, significant negative reactivity feedbacks will be generated due to the radial expansion of diagrids and pads. This negative reactivity feedback, together with other negative reactivity feedbacks induced by fuel axial expansion and the Doppler effect, can take over the positive reactivity feedback caused by coolant density decrease. Hence, core power will begin to decrease and eventually stabilize at approximate 1200 MWth (Fig. 2). Besides, the maximum temperatures of fuel and clad will also decrease and reach ~ 1400 ^{o}C and ~ 700 ^{o}C after 300 seconds into the ULOF transient (Fig. 3).

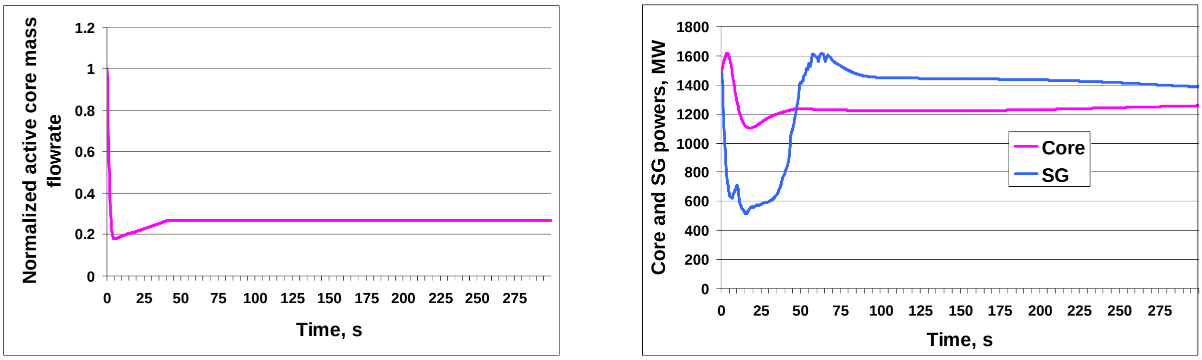


FIG. 2. Evolutions of primary coolant flowrate (left) and core/SG power (right) with time for EOC as calculated by TRACE/FRED (PSI)

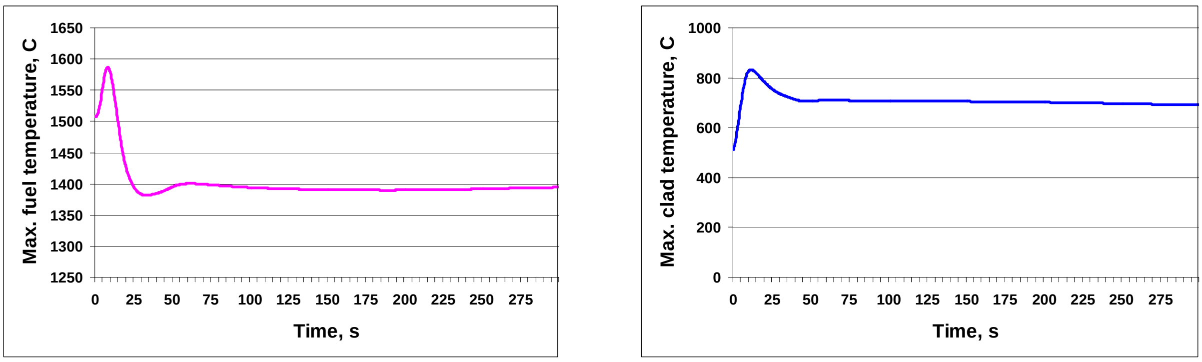


FIG. 3. Evolutions of the peak fuel (left) and clad (right) temperatures with time for EOC as calculated by TRACE/FRED (PSI)

Since SG power will be higher than core power after 50 seconds into the transient (Fig. 2), the maximum temperature of clad will continuously decrease. The highest temperatures that can be attained by fuel and clad of the peak fuel pin during ULOF are 1586 ^{o}C and 828 ^{o}C respectively, lower than their corresponding failure limits as minimum clad failure times of 10^{+5} seconds under the minimum coolant flow conditions (flow undershoot conditions) are calculated.

Based on the above observations one can state that the ELFR plant as designed can accomodate a ULOF transient.

4.2. Unprotected loss of heat sink transient (ULOHS)

The unprotected ULOHS transient is initiated by the loss of feedwater to all steam generators without reactor scram. The secondary circuits are automatically isolated and the DHR-1 system is activated (3 out of 4 IC loops are supposed to be in service).

The primary system remains in forced circulation, so the core mass flow rate does not significantly decrease during the transient (Fig. 4). The core power progressively reduces (Fig. 4) towards the decay level due to negative reactivity feedbacks introduced by the core temperature increase. The steam generator power (Fig. 4) in excess to DHR-1 power in the first 1000 s of the transient is caused by water vaporization and steam release to the atmosphere through the relief valves, because of the over pressurization of the secondary circuits in the initial phase, after their isolation and DHR-1 start up.

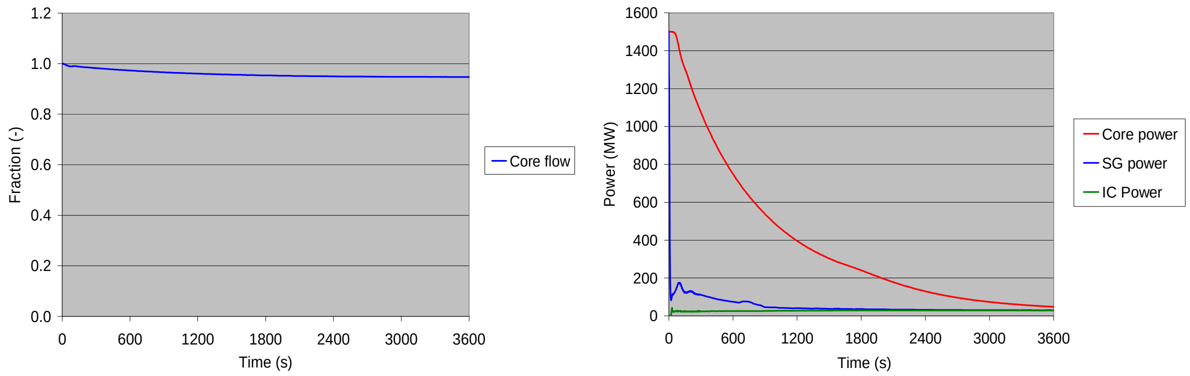


FIG. 4. Evolutions of core mass flowrate (left) and core/SG/IC powers (right) with time for EOC as calculated by RELAP5 (ENEA)

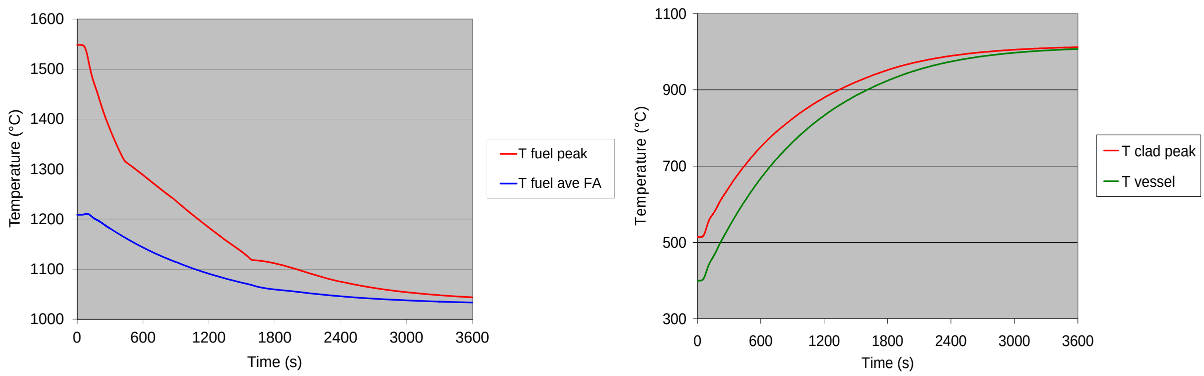


FIG. 5. Evolutions of peak fuel (left) and clad/vessel wall (right) temperatures with time for EOC as calculated by RELAP5 (ENEA)

The primary temperature progressively increases because of the strong mismatch between core power and heat removed by the secondary side of SG. At the same time, the ΔT through the core reduces according to the power decrease, leading to a practically uniform temperature distribution in the whole primary system. Also the vessel wall temperature reaches the equilibrium with the lead temperature in the medium term. As the power level decreases, the maximum fuel temperatures decrease close to clad temperatures (Fig. 5). After one hour, the level of temperature reached in the primary system is about 1000 °C, so that the integrity of the fuel rod clad cannot be assured from this point onward. However, the most challenging situation involves the vessel structure. Because of the very high temperature increase, the structural integrity of the vessel cannot be guaranteed in the long term.

4.3. Unprotected loos of flow and loss of heat sink transient (ULOF+ULOHS)

The unprotected ULOF+ULOHS transient is initiated by the simulataneous loss of primary pumps and of feedwater to all steam generators without reactor scram. The secondary circuits are automatically isolated and the DHR-1 system is activated (3 out of 4 IC loops are supposed to be in service). The results of these calculations are very similar to those of the ULOHS case as sufficient natural circulation (> 10% nominal) in the ELFR assures sufficient mixing of the coolant throughout the primary system. Again, high vessel temperatures cannot guarantee the long term structural integrity of the vessel.

4.4. Unprotected reactivity insertion transient (UTOP)

ELFR overpower transient at HFP and EOC conditions is reported in this section, namely: 260 pcm reactivity insertion within 10 sec time interval.

As can be observed in Fig. 6, an insertion of 260 pcm in 10 sec time interval at EOC conditions leads to a power jump of ~ 2.42 nominal. The maximum fuel and clad temperatures under EOC conditions increase from 1539°C and 513°C to 2677 °C (fuel in the fuel pellet center will start melting, but fuel melting will not progress to the fuel pellet surface) and 719 °C respectively (Fig. 6).

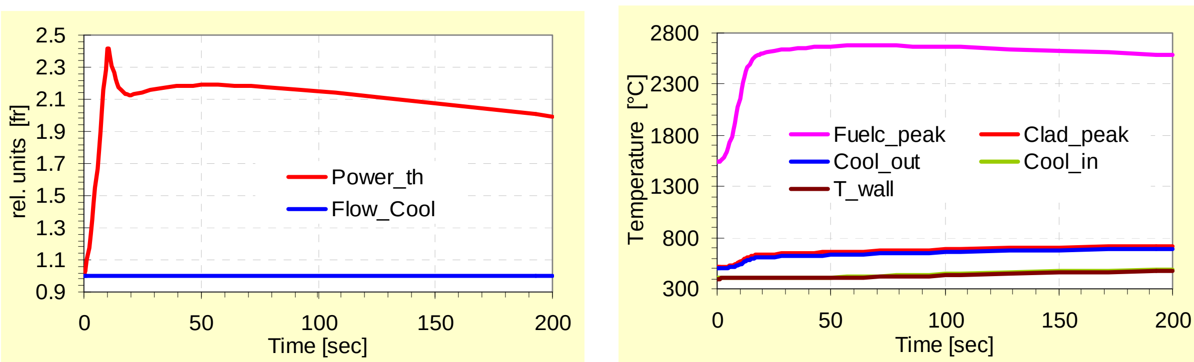


FIG. 6. Evolutions of core power and flowrate (left) and peak fuel/clad/coolant/vessel wall temperatures (right) with time for EOC as calculated by SIM-LFR (KIT)

From the performed analysis one can see, that ELFR reactor peak fuel pin cladding survives this transient, however local fuel melting should be expected in the center of the peak fuel pins (pellets).

4.5. Unprotected sub-assembly (SA) blockage transient

Two types of calculations were performed (for both cases nominal power at HFP, no reactor trip and no radial heat transfer are assumed) when analysing unprotected SA blockage transient:

Case 1: The flow area through the hottest SA is blocked 97.5% instantly at 1 sec transient time. Thus a flow area of only 2.5% remains open for the flow of coolant. Of interest here is time to clad failure after blockage initiation.

Case 2: The maximum clad temperatures of the peak pin as a function of the blockage area are determined.

The simulations were performed using SIM-LFR code. However it should be noted here that in both cases it was assumed that coolant flow in the blocked SA is linearly proportional to the SA blocked flow area.

For the 97.5% SA blockage transient at EOC, the peak pin will fail ~93 sec into the transient (transient initiation at 1 sec transient time) as the cladding temperature will reach 1015 °C (Fig. 7), with a peak pin fission gas pressure of ~41 bar.

As a result of the SA blockage, the flow rate will initially decrease to ~ 14 % nominal, gradually recovering to about 24% flow rate at ~50 sec into the transient due to changing SA pressure conditions. The power remains at 100% nominal throughout the transient (Fig. 7).

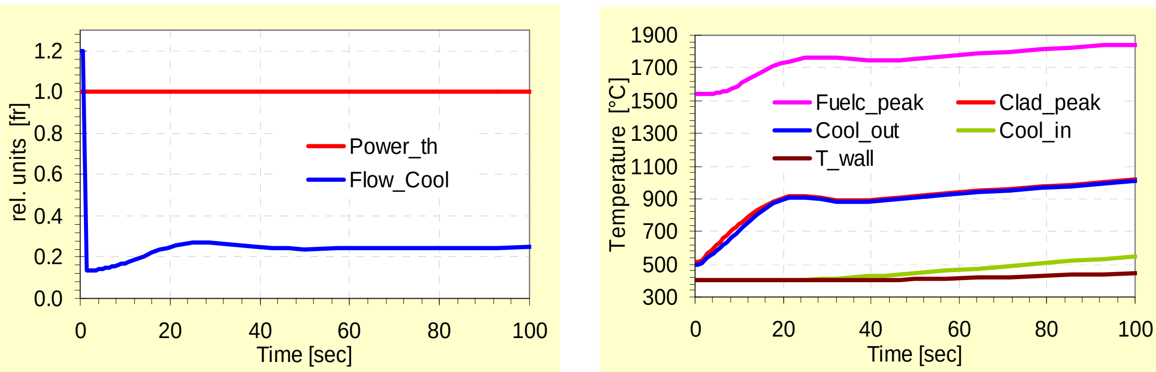


FIG. 7. Evolutions of core power and flowrate (left) and peak fuel/clad/coolant/vessel wall temperatures (right) with time for EOC as calculated by SIM-LFR (KIT); Case 1 (97.5% flow area blockage)

When determining the maximum clad temperatures of the peak pin as a function of blockage area – analyzing Case 2, several different cases were run at EOC conditions, varying the blockage area ranging from 20% to 97.5% (20, 40, 60, 65, 70, 75, 80, 90, 95 and 97.5%). Simulation results are presented in Fig. 8.

Following the performed analysis, it can be stated that:

(1) The ELFR will not experience any fuel pin failure for blockage areas less than 75%, even under unprotected conditions;

(2) For blockages above 75%, clad failures must be expected (Fig. 8);

(3) Fuel melting is not an issue for the ELFR. Fuel melting temperatures are not reached even in 97.5% SA blockage case.

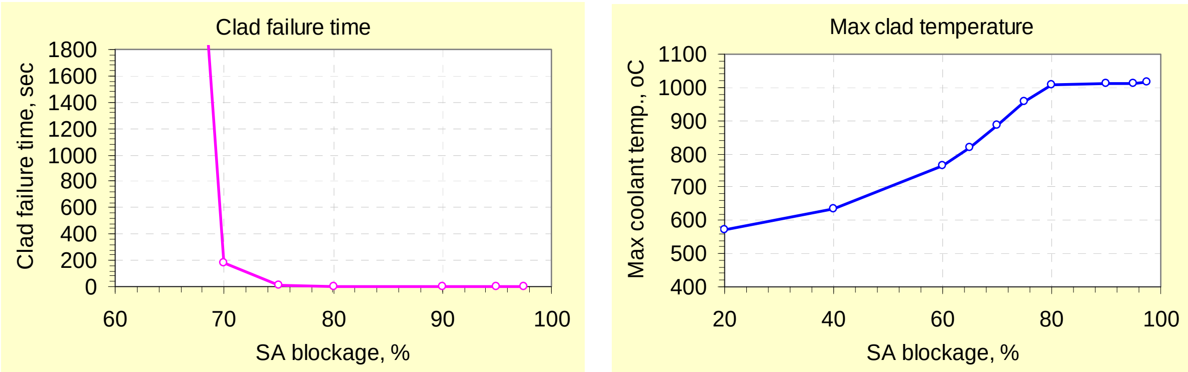


FIG. 8. Evolutions of clad failure time (left) and peak clad temperature (right) with time for EOC as calculated by SIM-LFR (KIT); Case 2

4.6. Steam generator tube rupture (SGTR) accident

The SGTR accident initiates with the postulated sudden rupture (induced by a variety of degradation processes, such as cracking, wall thinning, etc.) of one or more SG tubes. As a result, a water flow rate is injected into the hot lead (lead-water interaction - LWI) determining the resultant pressure peak due to water vaporization. The induced consequences of the SGTR tube rupture failure (e.g. production of high pressure steam bubbles, pressure wave propagation in the SG itself and/or in the whole primary system) are strictly dependent on the injected water flow rate and total inventory interacting with hot lead.

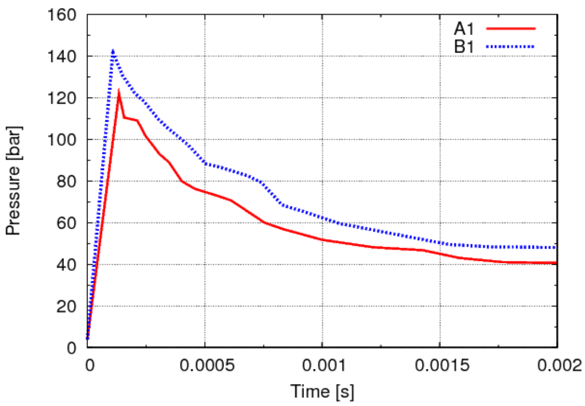
Two types of calculations have been performed with SIMMER III code: sub-series A and B, simulating respectively the single and double-ended guillotine rupture. In the latter case check valve failure was also assumed. Moreover water flow rate limiting mechanisms, such as the adoption of a Venturi nozzle, was placed inside each spiral tube to limit the amount of water injected (A3 simulation).

Basically, without any engineering safeguards, the initial interaction of the two fluids results in an instantaneous vaporization of injected water, responsible for the pressurization of the inner SG region. This is a very conservative hypothesis that needs experimental verification. In some reported experiments at higher pressures [3] only a small fraction of the injected water vaporized as most of the water (77.2% of the injected water mass) was transported in liquid form inside the steam bubbles to the cover gas region to be then separated into small droplets above the free lead surface.

The pressure peak (the duration of which is in the order of 10^{-4} s and thus not relevant from a mechanical point of view), close to the rupture location, reached ~140 bar in all the B sub-series and ~120 bar in all the A sub-series (Fig. 9) simulations, while the mean pressure in both cases is below 20 bar.

The pressure peak induced a lead displacement upwards in the SG pressurizing and compressing the cover gas, a second pressure peak (of about 35 bar in the A sub-series simulations) appears close to the upper SG plate.

The adoption of a Venturi nozzle allows limitation of the water mass flow rate injected into the lead and, in turn, the peak pressure due to LWI is greatly reduced. The maximum pressure close to the upper SG plate was less than 10 bar (Fig. 9), and the calculated mean value was about 3 bar.



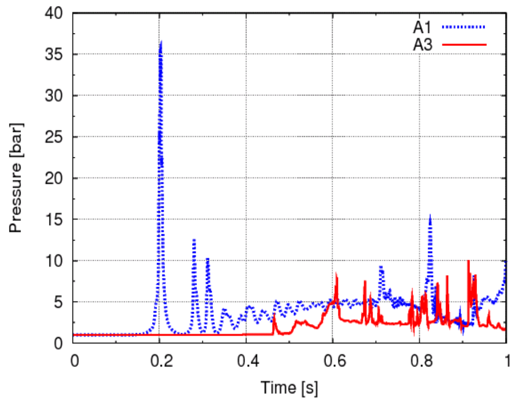


FIG. 9. First pressure peak inside the lead region close to the rupture (left) and pressure impulse on the upper plate (right), as calculated by SIMMER (CIRTEN)

Following this preliminary analyses, it can be stated that:

(1) without any engineering safeguards or limiting mechanisms (e.g. Venturi nozzle, safety valves on the upper side of the vessel, etc.), the LWI induced pressures, in case of instanteneous vaporization, could be severe enough to lead to structural risks for the SG itself (e.g. collapse of adjacent tubes), while it poses no likely threat for the integrity of the in-vessel structures;

(2) the vapour bubbles, generated during the LWI, have difficulty in reaching the core inlet section, as confirmed by the KALLA experimental data, under the action of the drag force produced by the liquid metal flowing downward in the downcomer region of the reactor (see also [3]).

5. CONCLUSIONS

In the framework of the LEADER project, the safety approach for a European Lead-cooled Fast Reactor (ELFR) has been defined and, in particular, all the possible challenges to the main safety functions and their mechanisms have been specified, in order to better define the needed provisions.

Safety analysis of the ELFR was performed testing all the “weak” points in the LFR design that were determined during the previous projects. The most important design basis condition (DBC) and design extension condition (DEC) transients were repeatedly analyzed for the re-designed ELFR configuration. The results of the safety analyses can be summarized as follows:

- Protected transients (PLOF, OVC and SLB): the automatic reactor shutdown activated by different scram signals is able to rapidly bring the ELFR plant to safe plant conditions. The consequent isolation of the secondary circuits and start up of decay heat removal system is able to maintain the plant in safe conditions in the medium and long term. In all transients, the potential of lead freezing in the coldest points of the primary system is reached after several hours into the transient, assuring sufficient grace time for manual, corrective operator action.
- Unprotected transients (ULOF; ULOHS and ULOF + ULOHS): due to the enhanced natural convection capability in the primary circuit, in case of ULOF the maximum temperatures reached in the primary system are low enough to assure the integrity of the clad and the vessel in the short term, providing sufficient grace time for corrective operator action.

  The main potential safety issue is the maximum reactor vessel wall temperature that might exceed 700 °C within ~12 min. The integrity of the clad and the vessel seems not guaranteed in the medium/long term, because of the high temperatures reached in the primary system. An optimization of the neutronic core design, in order to reduce the positive coolant expansion reactivity feedback could provide additional grace time.

- Reactivity insertion: for reactivity insertion of 200 pcm in 10 sec time interval at EOC conditions, peak fuel pin cladding survives and fuel melting is not observed, even in the center of the peak fuel pins (pellets). For reactivity insertion of 260 pcm in 10 sec time interval at EOC conditions, peak fuel pin cladding survives, however fuel melting should be expected in the center of the peak fuel pins (pellets). These transients envelope positive reactivity insertions of the Design Basis events such as fuel handling errors, control rods withdrawal or seismic core compaction.
- FA flow blockage: for blockages less than 75% blockage area, it is not expected any pin failures nor fuel melting, even under unprotected conditions. For blockage above 75%, peak power pins clad failure shall be expected, but fuel melting is not expected even for blockage over 97.5%. However there is time (several hundreds seconds) to detect the flow blockage occurrence, by means of temperature measuring devices installed at each FA outlet.
- SGTR accident: several limiting mechanisms and potentially important effects have been analyzed and suggest that: (i) the initial pressure shock wave poses no likely threat to in-vessel structures, except very few adjacent heat-exchange tubes; (ii) the sloshing-related fluid motion is well bounded in a domain beyond the heat exchanger; and yet (iii) the steam/water entrainment is expected to be comparatively limited due to the very large difference of density between steam and lead. The potential gradual pressurization of the vessel after SGTR due to inflow of the steam is limited by rupture disks to relief the resulting over-pressure. Moreover, a Venturi nozzle placed inside each spiral tube, mitigate the severity of SGTR interaction and reduce the potential effects on the entire reactor system. Anyway, a dedicated scaled facility should be foreseen to analyze in depth the SGTR phenomena further as part of the future R&D activities.

In order to assure prevention of freezing of the lead coolant at the coldest location of the primary loop, a tight and continuous operational control of the secondary coolant conditions is needed. Under certain adverse transient conditions it is conceivable that the primary lead HX outlet temperature (nominally 400°C, well above the freezing point of lead (327°C)) decreases to the feedwater inlet temperature 335°C, (only 8 °C margins to freezing). In addition, any malfuction in the FW temperature control could progressively bring the coolant to its freezing point.

In general, the safety analysis performed for the lead-cooled ELFR design demonstrated the extremely robust nature of this plant design when compared to other similar plant designs, ascribable to the inherently, large thermal inertia of the lead-cooled primary system and optimization of safety relevant control, safety systems and components.

Acknowledgements

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