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Methodology for the evaluation of fission gas release in lead fast reactor

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METHODOLOGY FOR THE EVALUATION OF FISSION GAS RELEASE IN LEAD FAST REACTOR

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Summary

The Lead cooled Fast reactor is a reactor design option selected within GEN-IV initiative. GEN-IV reactor concepts have to fulfill four strategic goals, namely: sustainability, economics, safety and reliability, proliferation resistance and physical protection. Fast spectrum nuclear reactors enable a much more reduction of uranium consumption. To increase the operability of the reactor, sufficient reactivity reserve has to be assured by the core designer. Hence the nuclear fuel has to be designed to withstand high burn up. Increasing the burn up the gas release issue become more and more important, thus a correct and reliable evaluation of fission gas release during normal operation and in case of accident is of a major importance. In the former case statistical analysis may complement more classical mechanistic evaluations; in the latter case different computational tools may be involved.

Best estimate methodology may imply use of different codes. Namely to perform a realistic assessment of the fuel behavior, especially in case of accident conditions, a chain of codes has to be adopted if an analyst wants to adopt a suitable tool per different disciplines.

1 Introduction

The Lead cooled Fast reactor is a reactor design option selected within GEN-IV initiative. GEN-IV reactor concepts have to fulfill four strategic goals, namely: sustainability, economics, safety and reliability, proliferation resistance and physical protection. Fast spectrum nuclear reactors enable a much more reduction of uranium consumption. To increase the operability of the reactor, sufficient reactivity reserve has to be assured by the core designer. Hence the nuclear fuel has to be designed to withstand high burn up. Increasing the burn up the gas release issue become more and more important, thus a correct and reliable evaluation of fission gas release during normal operation and in case of accident is of a major importance.

Best estimate methodology may imply use of different codes. Namely to perform a realistic assessment of the fuel behavior, especially in case of accident conditions, a chain of codes has to be adopted if an analyst wants to adopt a suitable tool per different disciplines.

2 Overview of Lead Fast Reactor

This section deals with the description of a reactor cooled by lead. The reactor is part of a European project called ELSY developed within the Generation IV framework.

The fast neutron nuclear reactor cooled by lead, or more briefly LFR (English acronym: Lead-cooled Fast Reactor) is a Generation IV fast reactor closed loop, which uses molten lead as a coolant or a eutectic lead (44.5 wt%) - bismuth (55.5 wt%), whose melting temperature is around 124 °C, considerably lower than the individual pure components, while the coolant boiling point is 1750 °C. Such a wide temperature difference allows the refrigerant to work at atmospheric pressure and rather high temperature, generally preserving a boiling margin up to 600 K. In addition high coolant temperature ensures a valid thermodynamic efficiency. By this kind of reactor it is also possible to produce hydrogen. Figure 1 shows a schematic representation of a lead cooled reactor.

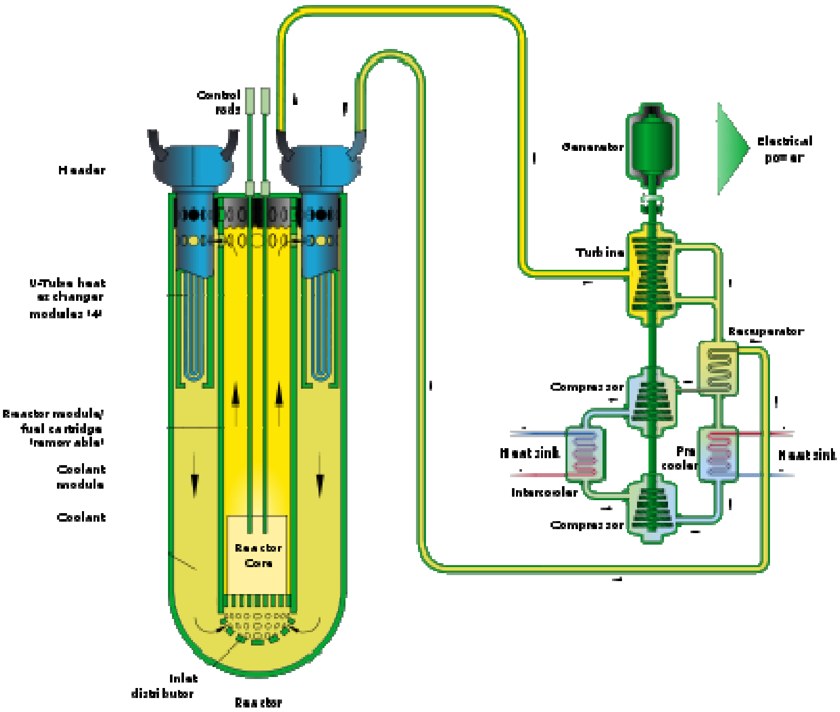

Figure 1 – Sketch of Lead Fast Reactor.

Within the European Project ELSY a consortium, composed by 19 participants and coordinated by Ansaldo Nucleare, is developing a LFR prototype. The ELSY project aims at demonstrating that it is possible to design a competitive and safe fast critical reactor using engineered technical features. The high lead density has the advantage that, in the hypothetical case of a core disruption, it is unlikely to lead to core compaction scenarios which might cause the insertion of large amounts of reactivity in a short time. The use of compact in-vessel steam generators and of a simple primary circuit with possibly all Internals being removable are among the reactor features for competitive electric energy generation and long-term investment protection.

The primary system concept studies carried out to date have led to several important design improvements resulting in a compact arrangement that features 8 innovative spiral tube Steam Generating Units, 8 Primary Pumps and 4 Decay Heat Dip Coolers installed in a vessel that is less than 9 m high, ref.[2].

The Reactor Vessel (RV) has a fixed cover that is basically a large annular steel plate with a central main steel upstand to accommodate the extended Cylindrical Inner Vessel (CIV). The fixed reactor cover plate incorporates penetrations which host the reactor components. The remaining inner part of the reactor cover is not conventional, because it consists of essentially the packed heads of the fuel/dummy assemblies that extend over the reactor cover plate. The cold collector is located in the annular space between the RV and the CIV (see Figure 2). The fuel assemblies are withdrawn from and plug into the core using a simple handling machine that operates in the cover gas at ambient temperature, under full visibility.

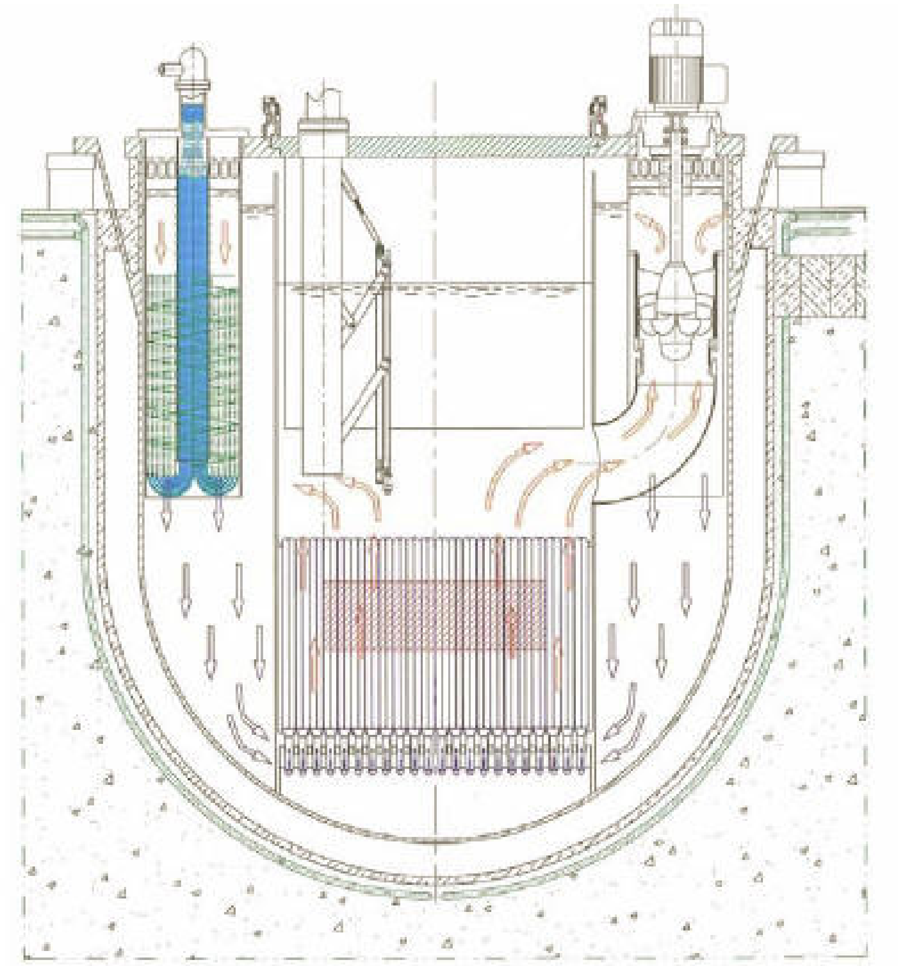

Figure 2 - Overview of the ELSY Reactor Vessel and main flow path

3 Overview of LFR fuel – focus on ESLY Project

The design of the European Lead-cooled System (ELSY) [1], is closely linked to GEN-IV initiative wherein Lead-cooled Fast Reactor (LFR) is amongst the six selected options. GEN-IV reactor concepts have to fulfill four strategic goals, namely: sustainability, economics, safety and reliability, proliferation resistance and physical protection. Fast spectrum nuclear reactors, such as ELSY, enable a much more reduction of uranium consumption. To reduce the operational cost it is important to reduce a number of the intermediate reactor shutdowns for the core reshuffling. Therefore the core composition should be designed with the sufficient reactivity reserve and small reactivity swing to assure at least two-three years of operation without fuel reloading or the core reconfiguration.

Desirable design objectives for a fast reactors such as ELSY include:

- a breeding ratio close to 1.0;
- a high specific power of the fissile material;
- high burn-up of fuel.

A Pu-enriched MOX with assumed 95% theoretical density has been adopted in the ELSY- 600 core pre-design round. The Pu isotopic vector is that of the reactor grade Pu extracted from the spent UO2 fuel of a typical PWR, discharged at 45 MWd/kgHM burnup and cooled down thereafter during 15 years. As for uranium, a depleted uranium isotopic vector, typical of industrial MOX production, has been adopted.

The choice of the cladding material and of the temperature range of the Pb-coolant operation is of the utmost importance for both the safety and the economics of the reactor. The margin to Pb solidification imposes to use the coolant temperature of at least 400 °C. The maximum coolant temperature is limited by strong corrosion of well-known cladding materials in Pb at temperatures higher than 550 °C. The available experience of using Pb and Pb-Bi eutectic coolants shows that the coolant bulk velocity has to be lower than 2 m/s, in order to avoid erosion problems during long-term operation in the temperature range from 400 to 550°C.

A fuel element residence time of 5 years, determined by corrosion limit, has been set as the target in ELSY. Two additional factors can limit the in-core residence time of the fuel element, namely the allowed fuel burnup and irradiation damage.

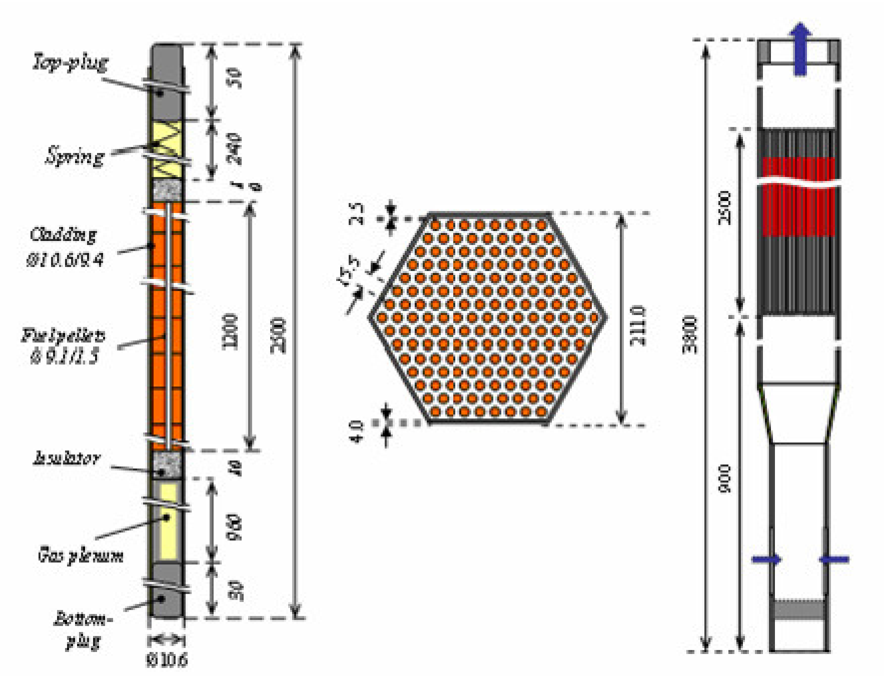

Figure 3 – Schematic view of fuel rod and sub-assembly

In Figure 3 a vertical schematic view of the proposed fuel pin and fuel assembly is shown. As a result of scoping neutronic calculations the active core height has been set to 1200 mm. The overall length of the needed gas plenum has been estimated to be close to that of the fuel column. This gas plenum space has been subdivided into two parts: 80 % in the lower *cold* part and 20 % in the upper *hot* chamber with a spring.

The first guess for the pitch (centre-to-centre distance between neighbor fuel rods) of the fuel rod lattice in the hexagonal bundle was derived from the thermal balance. It was thereafter optimized to ensure two additional constraints, namely a coolant bulk velocity ≤ 2 m/s and a pressure drop ≤ 0.1 MPa on an assembly. The number of the rods within the hexagonal bundle was fixed to 169. The overall length of the fuel assembly is 3.8 m.

The maximum burn up is mainly limited by the cladding resistance to the pressure of the released fission gas and to the fuel-cladding mechanical and chemical interaction.

4 Fission gas release

4.1 Actual status on phenomenon understanding

Fission gas production, migration and release in nuclear fuel has been studied since the earliest days of nuclear energy. At this time metallic fuels were in use and the early interest in fission gas release was mostly due to the large swelling they exhibited and the potential cladding damage due to enhanced gap pressure induced by the fission gas release. At that time theoretical analyses of fission gas behavior were quite simple, as was necessary for reactors in which fission density, burnup, and fuel temperatures were too low to produce the wide variety of mechanisms that are now recognized as occurring in modern highly rated fuels, ref.[3].

In general, a fission event entails – among others – two fission fragments that convey their kinetic energy to the fuel lattice. A fission fragment, close enough to a free surface (< 6-7 microns), can escape from the fuel due to its high kinetic energy (60-100 MeV). This is called fragment, a collision cascade or a fission spike with a stationary gas atom near the surface can also cause the latter to be ejected if it happens within a distance close enough to the surface. This process is called release by knock-out. Finally, a fission fragment travelling through oxide looses energy, causing a high local heat pulse. When this happens close to the fuel surface, a heated zone will evaporate or sputter, thereby releasing any fission product contained in the evaporated zone. Recoil, knockout and sputtering can only be observed at temperatures below 1000 °C, when thermally activated processes do not dominate. They are almost temperature independent and therefore called athermal mechanisms. It is generally of little importance in reactor at intermediate burnup levels. The fraction of athermal release is roughly under 1% for rod burnups below 45 MWd/kgU, and accelerates to roughly 3% when the burnup reaches about 60 MWd/kgU. Migration pathway for Xe atoms is quite complex. Xe is trapped at a uranium vacancy in UO2+x, at a tri-vacancy cluster in UO2-x and at a di- or tri-vacancy in UO_2. Since the local environment of the migrating Xe atoms is supposed to become the charged tetra-vacancy for all stoichiometries, the mechanism for diffusion only considers the association of a cation vacancy to the trap sites (Uranium vacancies as the slower moving species are rate-controlling for most diffusion related processes in UO2). The lattice diffusion coefficient is influenced by the temperature, deviations from stoichiometry and additives (e.g. Cr, Nb), phase changes and therefore also indirectly by the burnup. Also the fission fragments are assumed to contribute to the diffusion process, which is referred to as irradiation enhanced diffusion. This is due to the interaction of the fission fragments and the associated irradiation damage cascades with the fission gas atoms in the lattice, resulting in a displacement of the gas atoms. This effect dominates the diffusion process at temperatures below 1000°C and is temperature independent. For temperatures between 1000 °C and 1400 °C, vacancies necessary for the gas atom diffusion are assumed to be created both thermally and by the damage cascades related to fission fragments. Above 1400 °C, a purely thermally activated diffusion coefficient is applied, i.e. thermally created vacancies for diffusion are predominant. recoil release. When fission fragments make elastic collisions with the nuclei of lattice atoms, a collision cascade appears. The interaction of a fission.

In nuclear fuels, either natural (e.g. impurities, dislocation lines, closed pores, etc.) or radiation produced imperfections in the solid (e.g. vacancy clusters in fission tracks, fission gas bubbles, solid fission product precipitates, etc.) depress the amount of fission products available for diffusion by temporarily or permanently trapping the migrating atoms. Conducted experiments show that for burnups characteristic of power reactors, gas atom trapping due to (intragranular) fission gas bubbles in the grains is predominant. The trapping rate depends on the size of the intragranular bubbles, hence on temperature, fission rate and burnup. A second important effect of trapping occurs at grain boundaries. It deals with the delay for the onset of thermal fission gas release, via the bubble interconnection mechanism.