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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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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.

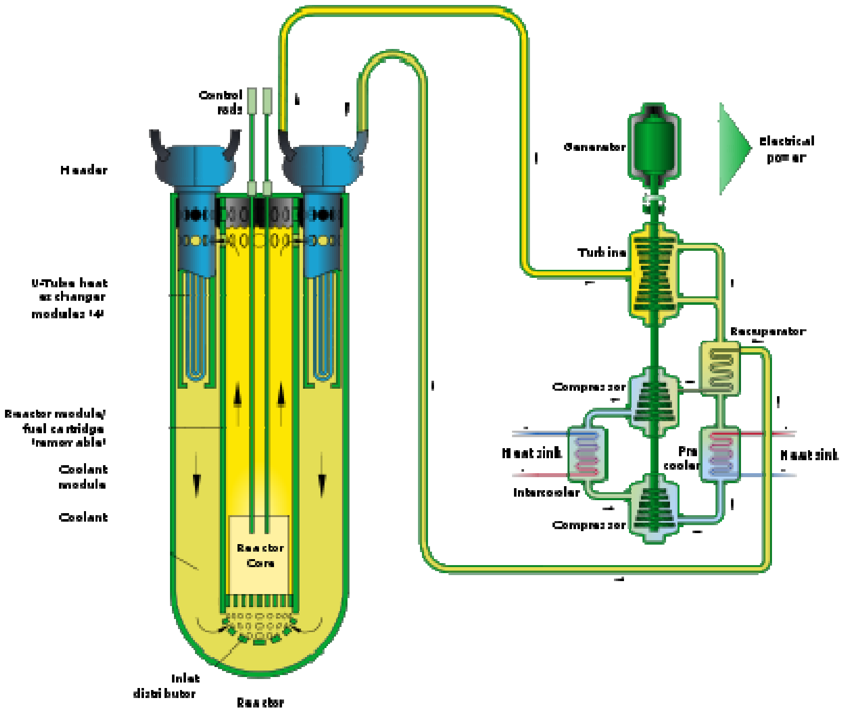

Figure 1 – Sketch of Lead Fast Reactor.

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.

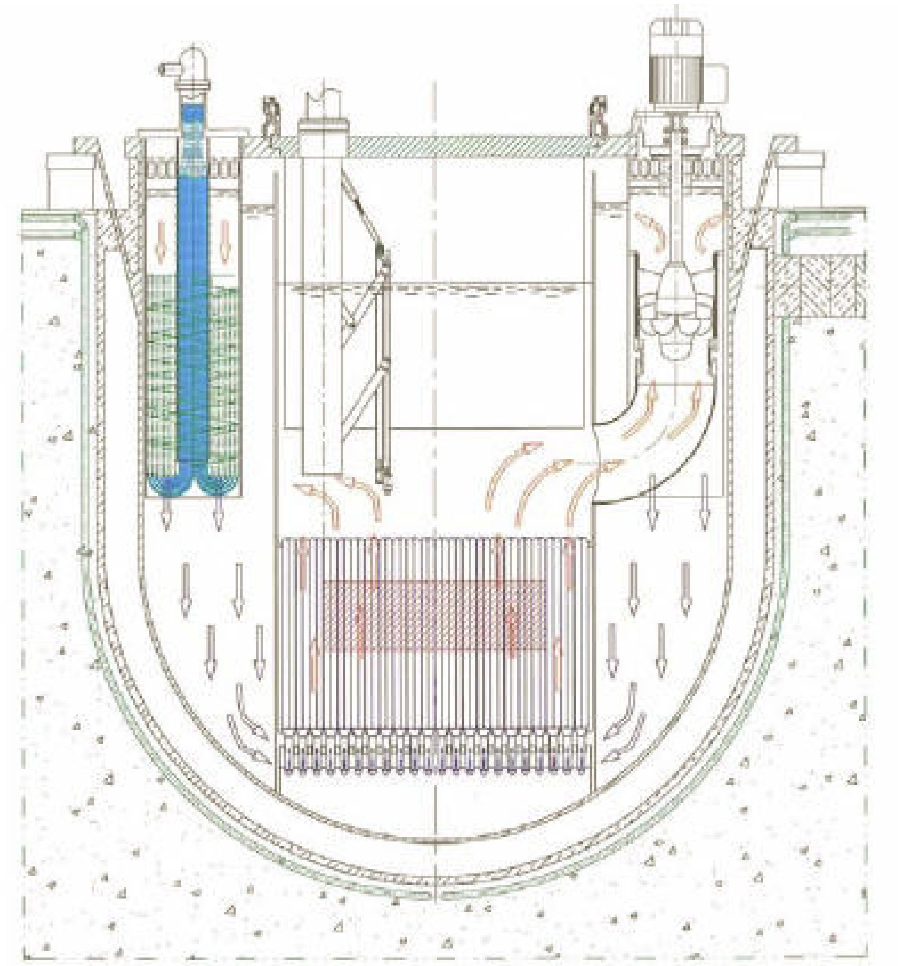

Figure 2 - Overview of the ELSY Reactor Vessel and main flow path

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.

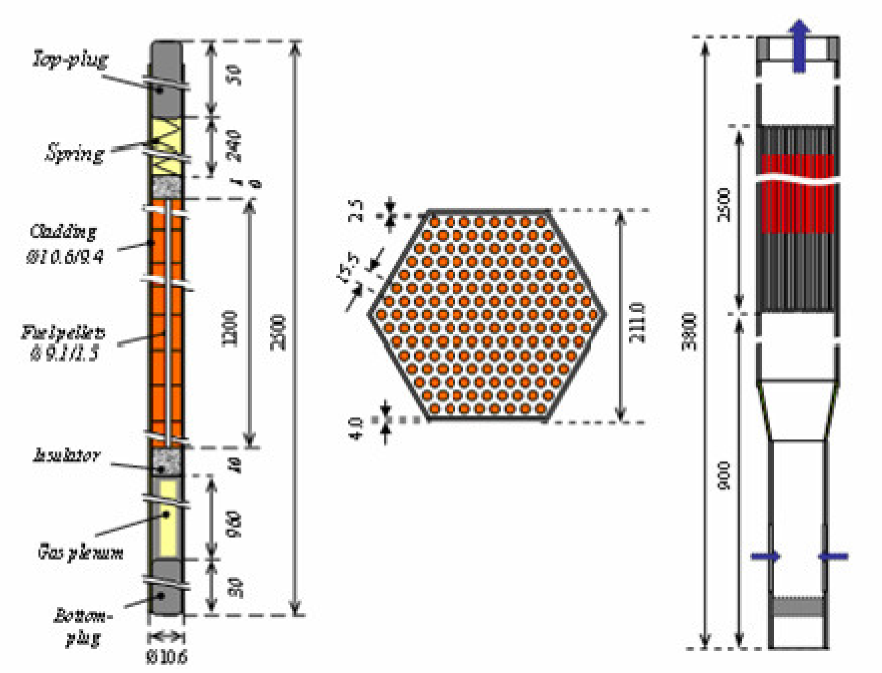

Figure 3 – Schematic view of fuel rod and sub-assembly

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.

A fraction of the gas atoms trapped in bubbles can be re-dissolved in the surrounding matrix through the interaction of a fission fragment with the bubble. Two different types of mechanisms are proposed to explain the experimental observations. On one hand, microscopic models consider the resolution of one gas atom at a time when interacting with a fission fragment or an energetic atom from the collision cascade. Macroscopic models on the other hand consider the complete bubble destruction, but there is still discussion about the detailed mechanisms. For (larger) grain boundary bubbles resolution is supposed to be less effective.

Grain boundary diffusion is the most commonly observed route for solute migration in polycrystalline materials. It is generally accepted that diffusion in crystalline solids proceeds more rapidly along grain boundaries than through the lattice. This is due to the atomic jump frequency in these planar defects which is about a million times greater than the jump frequency of regular lattice atoms in stoichiometric materials at 0.6 times the absolute melting temperature. Nevertheless, there is a switch from release assisted by grain boundary diffusion in trace-irradiated UO2 to trapping and eventual interlinkage of the intergranular bubbles. This switch occurs early in life, so that grain boundary diffusion is only considered to contribute to the precipitation of fission gas atoms in grain boundary bubbles, rather than to the long range transport along grain boundaries to the free surface of the pellets.

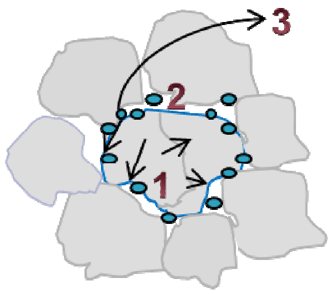

Figure 4 – Graphical representation of fission gas release

In the Figure 4 a graphical representation of the fission gas migration and release is shown. Three major steps can be recognized:

1. Fission gas atoms diffuse from within the fuel grain to the grain surface.
2. The fission gas atoms accumulate at the grain surface in gas bubbles.
3. When the surface is saturated with bubbles, the gas is released out of the fuel pellet matrix.

FGR strongly depends upon temperature and burn-up (time of in-core exposure). As an example FGR is <1% for temperatures below ~1000- 1200°C and rise up as high as ~10-20% at temperature of about 1500 °C.

Typically in fuel under nominal operating conditions, only normal grain growth is observed, i.e. large grains grow at the expense of smaller ones. Such a growth affects fission gas release in two ways. First of all, grain boundary sweeping provides another mechanism for the collection of gas at these internal surfaces from which release can occur. The collection results from the low solubility of the fission gas, hence the sweeping grain boundary does not redeposit any gas in the newly formed crystal behind it. The moving grain boundary acts as a fission gas filter. Secondly, the average diffusion distance for the fission products created in the grain increases. Unlike the first consequence this tends to reduce the release rate. Grain boundary sweeping occurs at temperatures above roughly 1600 °C.

The migration of fission gas bubbles provides an alternative to the sequence “bubble formation-resolution-gas atom diffusion” in order to describe fission product release from nuclear fuels. Migration of bubbles in the oxide fuels has two other important consequences, namely the columnar grain growth with the concomitant central void formation (observed in fast breeder reactor fuel), and the coalescence of the bubbles which gives rise to fuel swelling. Under normal operating conditions, however, fission gas bubbles remain small (typically below 20 nm) due to resolution, and show a small mobility at least up to 1800 °C. This is partly explained by the pinning, by dislocations and other crystal defects.

Fission gas bubbles appear along grain boundaries after a certain burnup, depending on the temperature history. When bubbles interconnect, they form a so-called tunnel network through which the gas can be released. The bubble interconnection is a reversible process, for the tunnel network can close again under the influence of the surface tension when the outgoing flux of gas atoms outweighs their supply. The bubble interconnection has two important consequences. First of all, it determines the onset of release as the release remains small (due to athermal release) before grain boundary bubbles interconnect with open grain edge tunnels. This incubation period is reflected in the Vitanza threshold for fission gas release. The ensuing release corresponds to a seepage process. Secondly, when grain face bubbles interconnect and form snake-like tunnels, there will be a sudden release of the gas accumulated in these bubbles, referred to as burst release. This can also be interpreted as a sudden interconnection or opening of grain face bubbles due to microcracking along grain boundaries during abrupt power variations. Cracking entails a sudden opening of a fraction of the grain boundaries with the instantaneous venting of the corresponding fraction of the accumulated gas atoms.

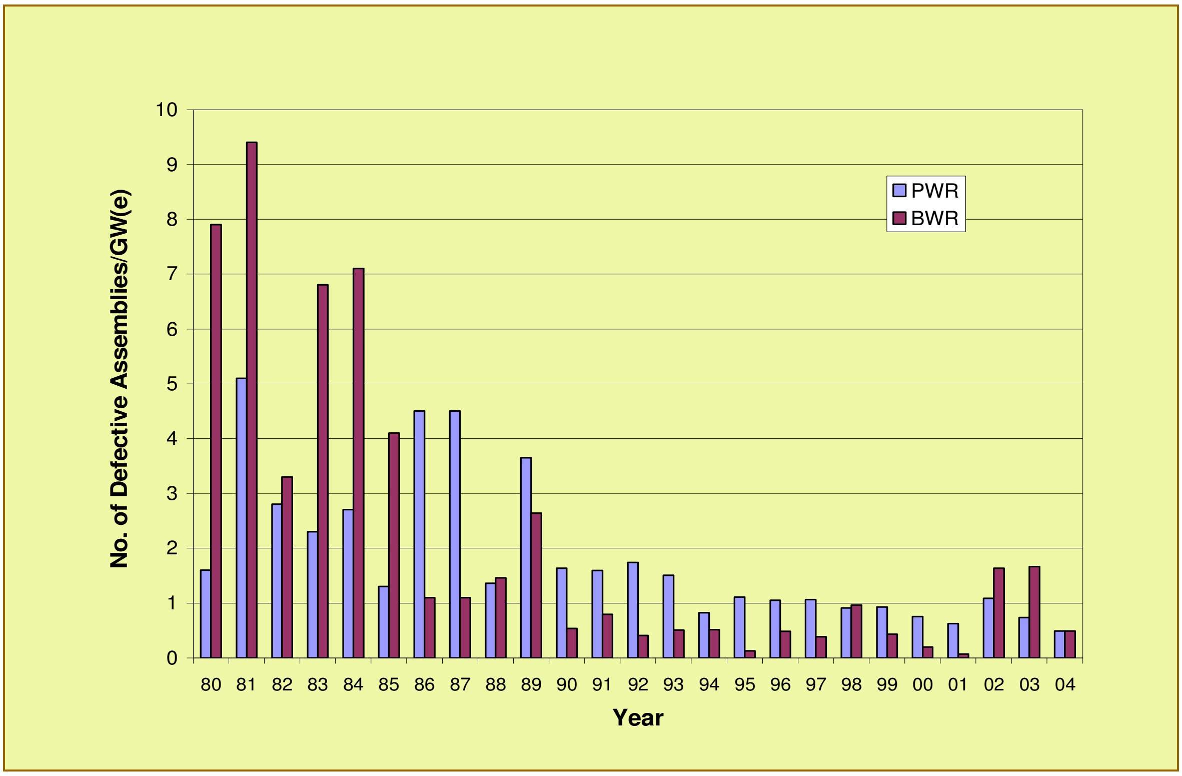

Figure 5 – Fuel failure trends

The in deep study of mechanism attained to the FGR prediction, have been conducted also for other aspects affecting the fuel performance and in general its integrity. This increasing of fuel behavior understanding is reflected also in the number of defective assemblies registered during NPP power production. Figure 5 reports the evolution of the defective fuel within a period of 25 years. Fuel can fail at typically 1 failure/reactor-year and there are main fuel failure that causes failure at normal condition, ref. [4]**:**

- Pellet-cladding interaction
- Debris fretting
- Rod vibration
- Degradation of failed fuel causing very large clad cracking
- Excessive corrosion (limited cases)
- Crud deposition (comes now and then, plant dependent)

4.2 Fission gas release related issues

In order to ensure the safe and economic operation of fuel rods, it is necessary to be able to predict their behaviour and life-time. The accurate description of the fuel rod's behaviour, however, involves various disciplines ranging from the chemistry, nuclear and solid state physics, metallurgy, ceramics, and applied mechanics. The strong interrelationship between these disciplines, as well as the non-linearity of many processes involved calls for the development of computer codes describing the general fuel behaviour. Fuel designers and safety authorities rely heavily on these types of codes since they require minimal costs in comparison with the costs of an experiment or an unexpected fuel rod failure. The codes are being used for R&D purposes, for the design of fuel rods, new products or modified fuel cycles and to support loading of fuel into a power reactor, i.e. to verify compliance with safety criteria in safety case submissions. In addition to steady-state irradiation, the fuel rod behaviour is also being simulated under transient and accident conditions, for example to assess the radiological source term.

For the simulation of off-normal operating conditions, specific "accident" codes require an estimation of the fuel rod status prior to the accident, which is often pre-computed by means of a code for normal operating conditions (even though there is no fundamental difference). In general, the uncertainties to be considered may be grouped into three categories. The first category deals with the prescribed quantities. The fuel rod performance code requires on input the fuel fabrication parameters (rod geometry, composition, etc.) and irradiation parameters (reactor type, coolant conditions, irradiation history, etc.). The second category of uncertainties is the material properties, such as the fuel thermal conductivity or the fission gas diffusion coefficients. The third and last category of uncertainties is the so called model uncertainties. A good example of such an uncertainty is the plain strain assumption in the axial direction as illustrated in Figure 6, representing the interaction of the deformed and cracked fuel with the cladding. Intuitively, it is clear that for a detailed analysis of such problems 2D or even 3D models are indispensable.

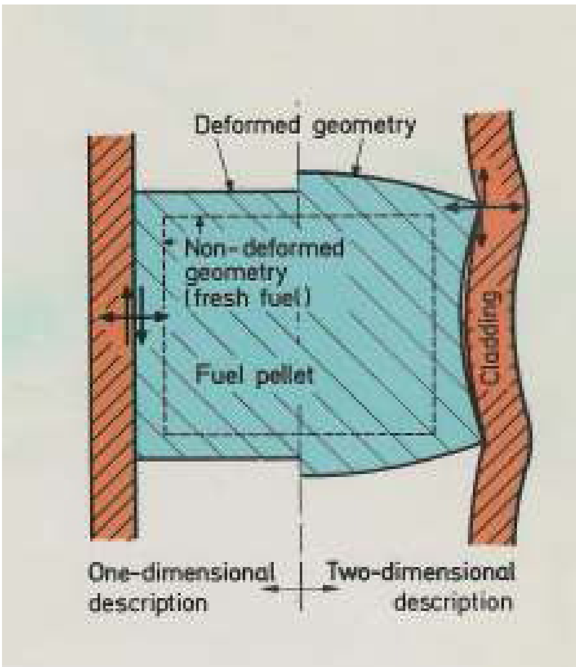

Figure 6 – Schematic view of a deformed fuel pellet; comparison between a one dimensional and a two-dimensional description.

4.3 Computational model and correlation

The present section provides a description of the fuel pin model by TRANSURANUS code. TRANSURANUS is a computer program for the thermal and mechanical analysis of fuel rods in nuclear reactors. It was developed at the Institute for Transuranium Elements (ITU). The TRANSURANUS code consists of a clearly defined mechanical–mathematical framework into which physical models can easily be incorporated. The *mechanical–mathematical concept* consists of a superposition of a one-dimensional radial and axial description (the so called quasi two-dimensional or 1 ½ D model). The code was specifically designed for the analysis of a whole rod. TRANSURANUS code incorporates *physical models* of thermal and radiation densification of the fuel, models of fuel swelling, fuel cracking and relocation, a model of generation of fission gases, a model of redistribution of oxygen and plutonium, and some other physical models. The code is exploited mainly by research institutions, industries and license bodies. Besides its flexibility for fuel rod design, the TRANSURANUS code can deal with a wide range of different situations, as given in experiments, under *normal, off-normal and accident conditions*. The time scale of the problems to be treated may range *from milliseconds to years*. The code has a comprehensive *material data bank* for oxide, mixed oxide, carbide and nitride fuels, Zircaloy and steel claddings and several different coolants. It can be employed in two different versions: as a *deterministic* and as a *statistical* code. The TRANSURANUS code was carefully designed to reflect the structure of the problem, which is defined by:

- the analysis of the fuel rod behavior at different times,
- the analysis of the different sections or slices at a specific time,
- the loop structure to obtain solutions of the various nonlinear problems,
- driver programs for the various options.

Consequently, the whole code is designed in levels:

- thermal analysis;
- mechanical analysis;
- fission gas release; and
- burn-up equations.

The basic assumptions and models can be summarized as follows:

Thermal analysis:

- Steady-state and transient analysis
- Phase changes included
- Advanced numerical solution technique (fast and stable)

Mechanical analysis:

- Constitutive equations

- Equilibrium
- Compatibility
- Superposition of one-dimensional radial and axial mechanical analyses.

This mechanical concept leads to a semi-analytical solution, which is solved by an effective numerical algorithm.

Physical models:

All important physical models are included, i.e. models for thermal and irradiation induced densification of fuel, swelling due to solid and gaseous fission products, creep, plasticity, pellet cracking and relocation, oxygen and Pu redistribution, volume changes during phase transitions, formation and closure of central void and treatment of axial friction forces.

The simulation of FGR requires the selection of two models: the first one deals with intra-granular gas behavior the second one refers to grain boundary behavior (inter-granular processes), ref. [6] and [7].

5 Methodology for evaluating the fission gas release

Key parameters driving the FGR are listed hereafter:

- LHR (Linear Heat Rate)
- Fast Neutron Flux
- Cladding temperature
- Pressure

The evaluation of such parameters requests a series of codes, which have to be coupled each other, in order to perform a state of the art best estimate analysis.

Depending on the reactor scenario to be considered (either normal operation or any accident condition), correct and reliable boundary conditions have to be taken into account. Among the other data the irradiation history (so called base irradiation) of the concerned fuel shall be evaluated.

The base irradiation is simulated by fuel pin mechanics code and through neutronic codes, in order to correctly estimate the effect of Burn Up on the behavior of the fuel (e.g. fission gas accumulation, fuel densification, etc.).

Power and temperature evolution are computed by TH 3D-NK coupled calculations. Subsequently they are used to provide boundary and initial conditions for the analysis performed by the fuel pin mechanic code. A "transferring map" is necessary to keep the correspondences among the NPP core configuration the TH model and the NK representations, [5]. Due to the radial power distribution with the reactor core generated by non flat neutron flux, best practices suggest that all channels have to be modeled.

Figure 7 highlights the connections among the different codes to be adopted in order to perform a complete evaluation of the fuel behavior, including fission gas release issue. In addition the main parameters exchanged between the different models are indicated, the name of codes recalled in the figure have to be intended as examples of suitable code. However a complete list of available computer codes per each technological area is out of the scope of the present report. In the chain of codes indicated in Figure 7 the evaluation of dose calculation is included to cover all the steps necessary in case of reactor safety assessment.

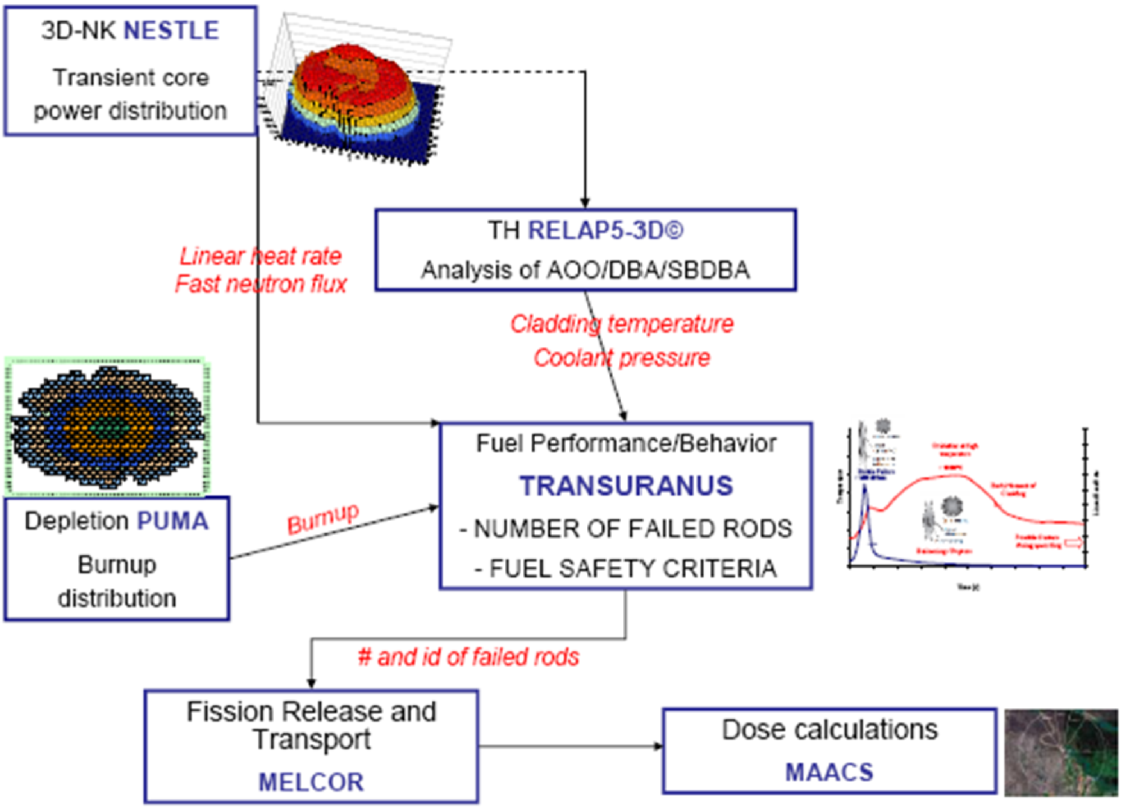

Figure 7 – Example of interaction between codes and associated main computed parameters.

5.1 Reactor nominal operation

During normal operation the fission gas generated during fuel irradiation, is confined within the rod itself. Namely gap, lower and upper plenum are ad hoc design to accommodate the predicted fission gas. Volumes of the three mentioned regions may differ among different fuel fabricants. Moreover even if a fuel rod design rational can be commonly followed, expected fuel behavior can be different because of different computational tools and assumptions. Other major source of prediction spread is the material properties which cannot be easily and very precisely modeled by simple equations.

To account for all the possible working conditions which the fuel may face during its irradiation history inside the core, the analysis approach has to be moved from the mechanistic to the statistic field.

The widely used mechanistic approach leads in many cases to discrepancies between theoretical predictions and experimental evidence indicating that most of the physical processes encountered are not sufficiently understood and cannot be modeled exactly. Some of these processes may even be of stochastic nature. Generally, the material behavior is far too complex to be characterized by models or correlations consisting of a few parameters. Therefore, for a better understanding of the uncertainties involved and their technological consequences, the influence of the various parameters needs to be investigated, which is best done by probabilistic methods, [8].

The probabilistic method Monte Carlo is applicable with a tool available in TRANSURANUS code. The Monte Carlo is used to provide perturbations of a number of input parameters which the code user selects based on their relevance in the concerned analysis. If one focuses on FGR issue, variations should be tested on data or physical properties which may affect the key parameters listed above, e.g. fuel thermal conductivity, neutron flux, pressure, fuel porosity, fuel density, etc.

5.2 Accident conditions

As stated above a complete analysis, especially if performed following a Best Estimate (BE) approach, should be conducted by:

- Realistic thermalhydraulic model of the reactor.
- 3D neutron kinetic core model.
- Performing coupled calculation, neutronic and thermalhydraulic.
- Evaluation by a specific fuel analyses code (e.g. TRANSURANUS) of FGR specific rate.

Each computational tool involved in the concerned evaluation needs an input deck and suitable boundary conditions. Being the code coupled (either internally or sequentially) a robust data transfer shall be developed. In addition major importance have to be put to the qualification issue. Namely not only the single input deck and code have to be properly qualified, but the whole the process too, [9]. Proof of qualification have to be provided also for the coupling technique especially if a very specific programs (i.e. code user made) are adopted.

Main classes of transient which are expected to have an impact on the key parameters influencing the FGR in a lead cooled reactor are: Reactivity Initiated Accident (RIA) and Unprotected Transient (equivalent to Anticipated Transient Without Scram in light water reactor technology).

Both transients are characterized by strong neutron flux variation, hence linear heat rate are expected to experience strong deviation from nominal conditions. Consequently, fuel and clad temperature are affected, increasing with a rate dependant from the severity of the postulated accident. The pressure inside the fuel rod changes too due to the strong variation in fuel properties and to the increase number of fissions.

Depending upon the postulated scenario, it may ends up with clad failure. It is worth to note that in case of cladding damage, fission gases are dissolved directly into the coolant. Hence the fuel analyses code must also be used to evaluate the number of pins damaged due to the postulated transient.

6 Conclusion

In the present report a methodology for the evaluation of fission gas release in lead cooled fast reactor has been presented. Such an evaluation can be performed either in normal or accident conditions. To conduct a complete analysis statistical methods exploited by Monte Carlo application, can supplement more classical mechanistic fuel performance evaluation.

For the purpose of the evaluation of fission gas release, key parameters influencing FGR have been recognized, namely:

- LHR (Linear Heat Rate)
- Fast Neutron Flux
- Cladding temperature
- Pressure

To perform a realistic assessment of the fuel behavior, especially in case of accident conditions, a chain of codes has to be adopted. This is a method that can be applied to conduct an analysis especially if a best estimate approach is followed. The drawback is that suitable level of qualification of input decks, codes and coupling technique (including data exchange) has to be proven.

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Curriculum Scientifico del Gruppo di Lavoro

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