PRELIMINARY ANALYSIS OF ALFRED UNDER SEISMIC SCENARIO

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Abstract

In the framework of the Horizon Europe ANSELMUS project, several numerical activities of accidental scenarios will have been performed in order to support the safety assessment of advanced HLM cooled reactors. A particular accidental scenario selected in this project is core compaction caused by earthquake. This paper presents the results of this accidental transient showing ALFRED behaviour in the event of an earthquake. The accidental transient was simulated by means of thermal hydraulic system code RELAP5/Mod3.3. The nodalization scheme is described in detail as well as transient boundary conditions and timing of events. This numerical activity showed that, despite the insertion of reactivity due to the core compaction caused by the earthquake, the temperature peaks of cladding and fuel are still below the safety limits and that the reactor reaches a new equilibrium point in a short time.

Keywords: *ALFRED, RELAP5, Nuclear Safety, Neutron Kinetic*

Introduction

The Advanced Lead Fast Reactor European Demonstrator (ALFRED) represents a cutting-edge nuclear reactor concept that holds significant promise for the future of sustainable energy generation. ALFRED is designed as a lead-cooled fast reactor, characterized by its advanced features and innovative technology. Its primary purpose is to demonstrate the feasibility and effectiveness of lead-cooled fast reactors in meeting the growing global demand for clean and reliable energy. Developed as a collaborative effort within the FALCON Consortium, ALFRED aims to address key challenges in nuclear energy, including enhanced safety, reduced environmental impact, and efficient resource utilization [1]. A staged approach summarized in Table1 has been put forward for the ALFRED reactor to progressively gain the operating experience and face validation issues [2].

Table. 1. ALFRED main parameters in the stages of operation

Parameter	Unit	Stage 0	Stage 1	Stage 2	Stage 3
Core inlet temperature	° C	390	390	400	400
Core outlet temperature	° C	390	430	480	520
Core thermal power	MW	0	100	200	300
Steam line pressure	bar	/	170	175	180
Steam line temperature	° C	/	420	435	450

The concept involves commencing operations at minimal power and temperature, gradually ramping up both using the reactor itself to validate subsequent operational phases and the concurrent advancement of the R&D program. Three power levels are anticipated (100, 200, and 300 MWth), at constant primary flow rate and increasing temperature. A necessary first step prior to this requires numerous R&D endeavours.

As reported in [3], previous neutron and thermal-hydraulic analyses have been carried out on previous configuration of ALFRED LEADER version. The simulated accidental transients are the UTOP (Unprotected Transient of Over-power) and the ULOOP (Unprotected Loss Of Off-site Power). The former, a reactivity insertion of 250 pcm in 10 seconds was considered and the mass flow rate in the primary loop is set constant at the nominal value. The latter, the loss of power to the primary pumps and consequently the loss of coolant flow rate in the core are assumed. For both transients, the failure of the SCRAM is expected. It should be noticed that the trend of the main quantities reported in this paper is consistent with the trends of both transients since the events of core compaction are the same of the UTOP (reactivity insertion) and of the ULOOP (the loss of flow in the primary).

Within this framework, the European projects ANSELMUS (Advanced Nuclear Safety Evaluation of Liquid Metal Using Systems) aims to address key challenges associated to LFR technology and seeks to design and demonstrate innovative solutions to enable the efficient and reliable operation of LFRs [4].

The first step is realised by performing a PIRT (Phenomena Identification and Ranking Table) on ALFRED. The PIRT aims to prioritise R&D activities, focusing on those aspects with lower level of knowledge and higher impact in terms of safety consequences, in relation to the requirements of the EURATOM safety directive. The starting point of the PIRT analysis is the ALFRED reference design and the selection of accidental transients to deepen the state of knowledge and the importance of certain quantities known as Figure of Merit (FoM). Open points may arise due to lack of evidence from available analyses. Thus, a full development of the PIRT requires a set of numerical simulations using different computer codes. Among the accidental events proposed in the framework of ANSELMUS European project, this paper focuses on the results of core compaction following a seismic scenario. A detailed nodalization scheme has been developed using RELAP5/Mod 3.3 system code, and the main results are discussed.

Description of the numerical model

Fig.1 shows the nodalization scheme of ALFRED implemented with RELAP5 system code used to simulate the transient of core compaction.

Although ALFRED is a 3-loop reactor, the model simulates the reactor with the 3 loops collapsed into a single loop according to code best practices. All control volumes coloured blue represent ALFRED reactor coolant system, while those coloured green represent the in-containment secondary loop between isolation valves. In addition, in the numerical model there is also ALFRED' DHR represented with components from c420 to c434 with the control volumes coloured yellow representing the pool side of the isolation condenser of the ALFRED DHR. However, as it is described later, the accidental transient is unprotected, so neither SCRAM nor the activation of this safety system are foreseen.

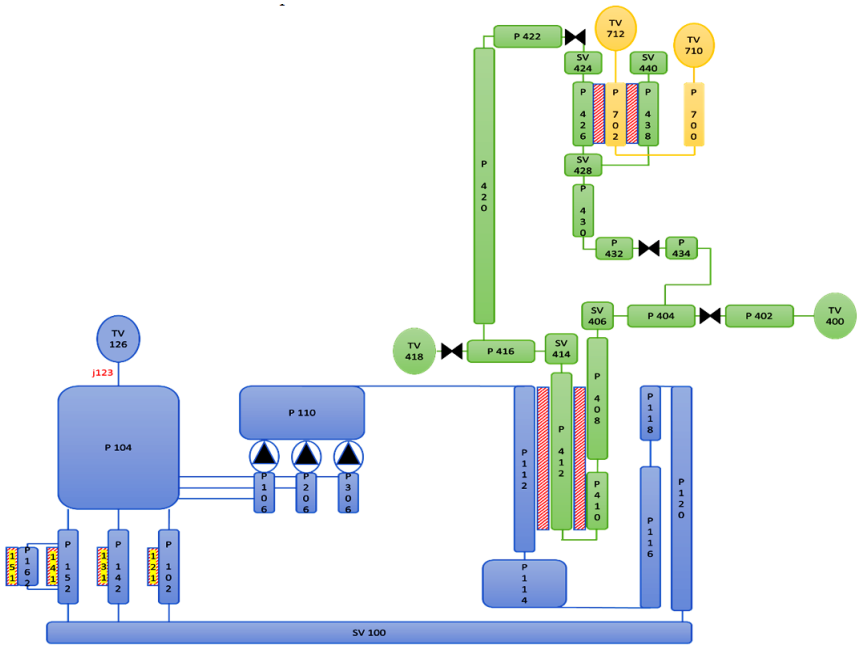


Fig. 1. R5 Nodalization scheme

ALFRED primary system model consists of the core, three pumps, and several pipes components that model the flow of the coolant according to [1]. ALFRED core is constituted by 134 Fuel Assemblies (FAs), 12 Control Rods (CRs), 4 Safety Devices (SDs) and one special position for in-pile irradiation experiments. The resulting 151 assemblies, composing the active region, are surrounded by 102 Dummy Assemblies (DAs), organized in two concentric rings, with the twofold purpose of reflecting leaking neutrons back to the core (inner ring) and of shielding the inner vessel from radiation (outer ring) [5]. In RELAP5 nodalization, the core is modelled as follows:

- C102 represents all subchannels that interface with the DAs
- C142 represents all subchannels that interface with the FAs except that of the hottest FA
- C152 represents all subchannels that interface in the hottest FA
- C162 represents the subchannel that interfaces into the hottest fuel pin

Each of the three regions has its own heat structures in which the material properties of both DAs and FAs are implemented. Particular attention should be paid to the fact that, since the transient to be simulated involves a several neutron phenomena, the neutron kinetics model of RELAP has been activated in the model. The input for the neutron kinetics model requires a set of reactor physics data such as delayed neutrons fraction, prompt neutron generation time and feedback coefficients (e.g., fuel temperature, coolant temperature, radial/axial expansions of structures). These quantities were furnished by ENEA, which designed the ALFRED core [5].The secondary loop involves the feedwater from the Bo Pentering the inlet c402 and the fluid exits as superheated steam on its way to the turbines. Finally, the primary and secondary loopsare thermally coupled by the bayonet tube steam generator which has been represented by the components ranging from c406 to c414 on the secondary side and from c112 on the primary side.

Transient analysis

The accidental transient starts from a steady-state numerical simulation representative of the normal operation of ALFRED in Stage 2. Therefore, the core generates a fission power of 200 MW which is removed by a flow rate of liquid lead of about 17000 kg/s. The coolant enters the core at a temperature of 400 °C and exits at 480 °C. Subsequently, the power is transferred to the secondary loop via the steam generator. This power is removed from the secondary loop with a flow rate of about 133 kg/s. The fluid enters at a temperature of 335 °C and exits at 435 °C. Stage 2 requires the secondary to be at a pressure of 175 bar. Table 2 summarizes the main results.

Table. 2. Full power calculation: steady-state results

Parameter	Unit	Value	Reference
Total lead mass flow rate	kg/s	16845.19	16845.60
Core inlet temperature	° C	400.7	400
Average core outlet temperature	° C	482.6	480
Maximum cladding temperature	° C	503.6	/
II SG inlet temperature	° C	335.1	335
II SG outlet temperature	° C	433.8	435
II SG mass flow rate	kg/s	132.9	132.9

Starting from the full power steady-state conditions, transient simulation is performed.

The starting point of the transient is the occurrence of the reference earthquake. Subsequent acceleration and deformation of the support structures of the reactor transmit through the reactor vessel to the internals, ultimately affecting the core by compacting it, resulting in a reactivity insertion. If we assume that the deformation remains within the elastic range due to the dynamic forces induced by the earthquake, the compaction of the core, and thus the reactivity, is expected to vary over time. To simplify without introducing unnecessary complexities for the PIRT objectives and keeping the results conservative, the reactivity insertion is linearized based on the maximum estimated compaction ($\Delta\rho$) and the time taken to achieve it ($t_{\Delta\rho}$). The transient starts at time t=0. Within 10 seconds, a uniform insertion of 250 pcm of reactivity occurs throughout the core. The forced circulation of lead within the Reactor Cooling System (RCS) ceases. Instead, forced circulation persists within the secondary loop, with no regulation of the feedwater flow rate, which remains constant at its nominal value.

To analyse a transient under conservative conditions, it is assumed that the accident is unprotected with both SCRAM and ALFRED DHR failure. The simulation of the accidental transient is performed for 5000 s but, as the following plots will show, as a result of the external perturbation due to the earthquake, the reactor reaches a new equilibrium point approximately in 2000 s.

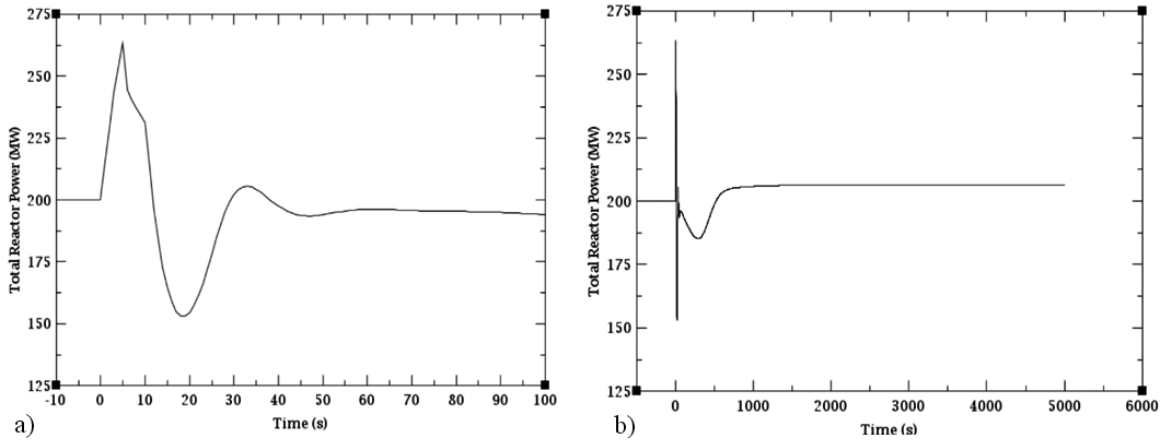


Fig. 2. Reactor power in short- and long-term phase

Fig. 2 shows the trend of the thermal power generated in the core. The figure on the left shows the initial value at full power corresponding to ALFRED Stage 2. Subsequently, due to the insertion of reactivity (0.75 $ in 10 s) the power rises sharply, reaching a peak of about 260 MW in 5 sec. After 5 seconds, although the reactivity insertion is still in progress, the thermal power generated in the core begins to decrease. This decrease in power is caused by the negative feedback coefficient of the fuel temperature: the abrupt increase in power causes a significant increase in the fuel temperature leading to a high insertion of anti-reactivity causing the decrease in power. Finally, the plot shows the new equilibrium power, which stands at 206 MW.

Fig. 3 shows the core inlet (black line) and outlet (red line) temperature trends of the coolant. It is noted that the core outlet temperature increases about 205 °C in 20 s, reaching a peak of 685 °C and settling at the new equilibrium temperature of 640 °C. The core inlet temperature remains at Stage 2 nominal value at 400 °C for 200 s transient and then decreases reaching equilibrium at 370 °C due to secondary loop continuous cooling.

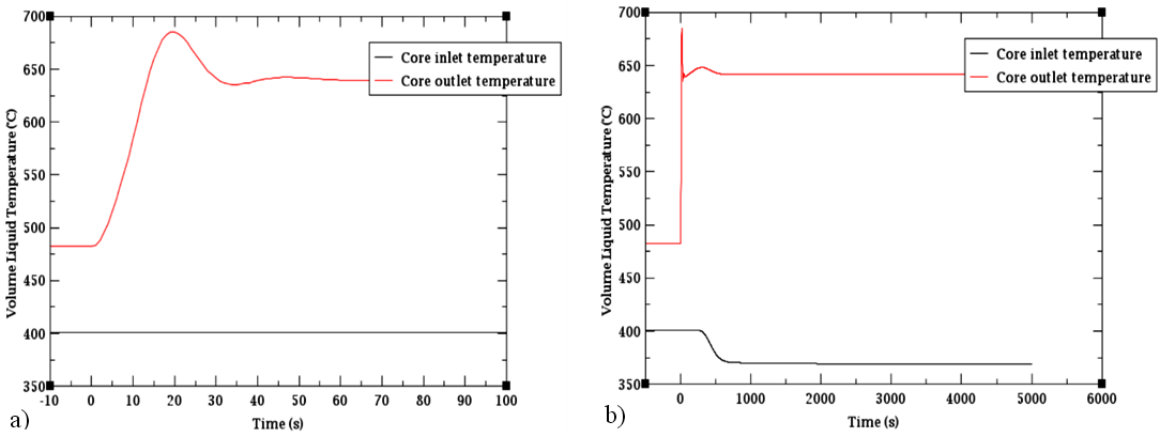


Fig. 3. Core inlet and outlet temperature of lead in short- and long-term phase

The trend of the primary coolant mass flow rate is shown in Fig. 4 where, due to the loss of the pumps, the flow rate decreases following the pumps coast down settling at about 5000 kg/s due to natural circulation (about 30% of nominal value).

Finally, the maximum fuel temperature and the maximum cladding temperature, respectively, selected as FoMs are shown in Fig. 5. The figure on the left shows the nominal temperature at full power of Stage 2 of 1400 °C and which has the same qualitative trend as that of the thermal power. The peak value of about 1650 °C and a new equilibrium temperature of 1560 °C are shown. Moreover, it is noticed that, according to [6], the peak temperature of the fuel is less than the melting temperature. Therefore, there seems to be no fuel melting during the transient.

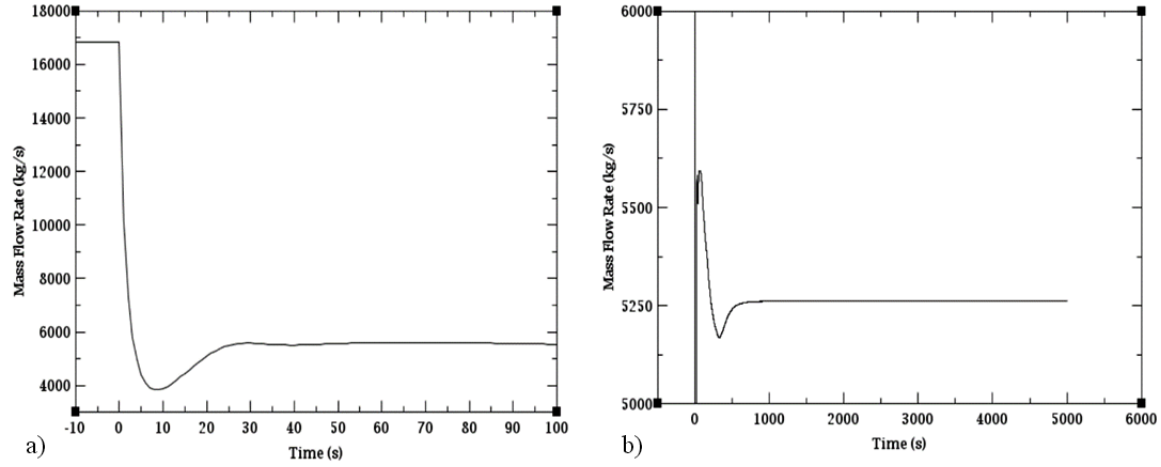


Fig. 4. Mass flow rate in the primary loop in short- and long-term phase

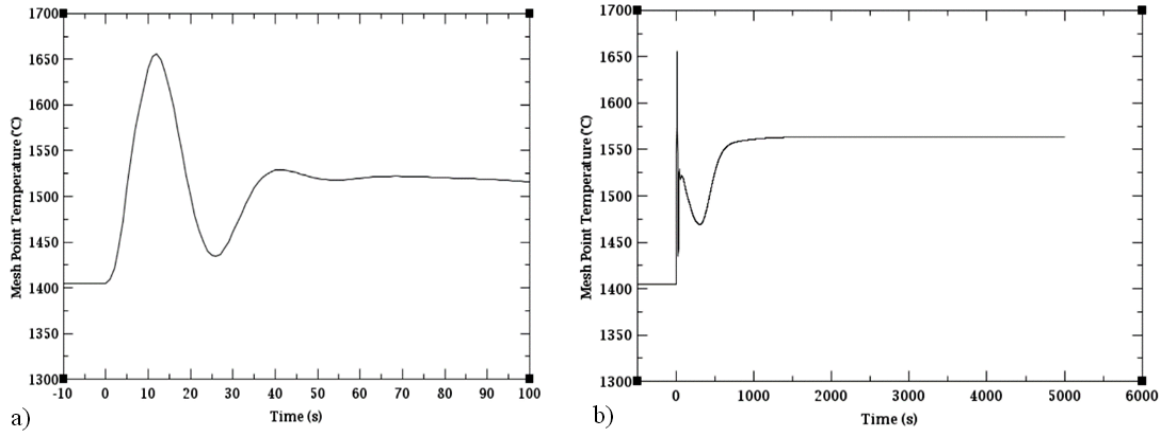


Fig. 5. Maximum fuel temperature in short- and long-term phase

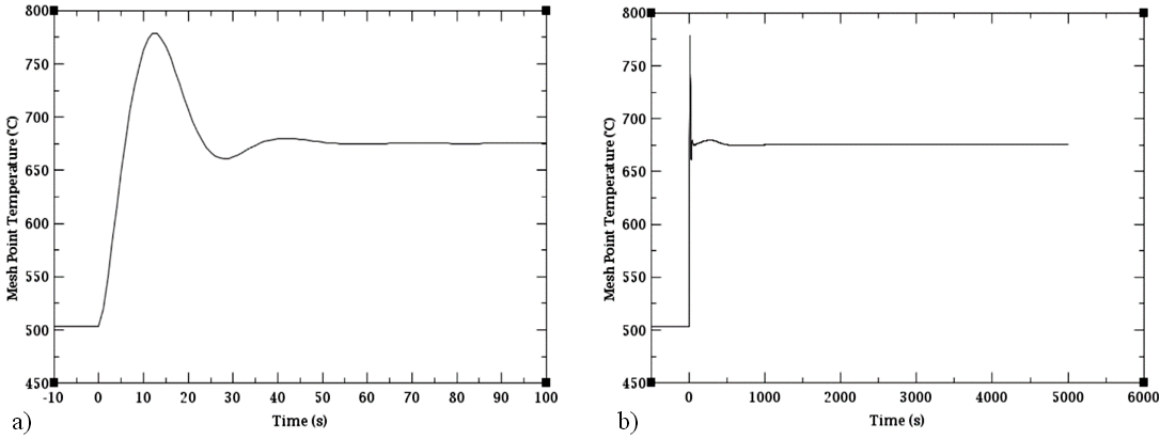


Fig. 6. Maximum clad temperature trend in short- and long-term phase

Fig. 6 shows the trend of the maximum clad temperature, being the second FoM to verify the integrity of the second barrier (the clad). It can be seen that it also has the same qualitative trend as the fuel temperature. It reaches the peak at about 778 °C in 12 sec of transient. In addition, it can be deduced from the graph that the final equilibrium temperature is 675 °C.

Finally, according to [7], the peak temperature of the clad is less than the melting temperature. Therefore, there is no clad failure due to melting during the transient. Even though additional mechanical evaluations should be performed to ensure of other failure phenomena absence, a first analytical evaluation can be performed by computing the time-to-rupture of the cladding with the following correlation [8]:

$$t_{failure} = 10^{\left(\frac{1520-\sigma_{eff}}{0.086 \cdot T} - 13\right)}$$

where σ_{eff} is the effective stress in the cladding expressed in MPa, T is the clad temperature expressed in K and $t_{failure}$ is the time-to-rupture expressed in hours. Given the peak temperature of the clad in this transient, it is estimated that the time-to-rupture of the clad is to be about 1060 hours.

Conclusions

This paper presents a preliminary 0-D neutron kinetic and thermal-hydraulic analysis of ALFRED reactor during a seismic scenario, assuming that such an initiating event causes a postulated core compaction which in turn causes a reactivity insertion. The work was carried out in the framework of the European project ANSELMUS. The proposed analysis has been performed by means thermal-hydraulic system code RELAP5/Mod3.3.

It should be noted that, to remain as conservative as possible, the following hypotheses have been assumed:

- The accident starts with a very high reactivity insertion (about 0.75 $, condition close to the prompt-criticality).
- The accidental scenario is unprotected, meaning that the control rods insertion (SCRAM) is expected to fail.
- The forced circulation in the primary loop is lost (conservative assumption).
- The mass flow rate of the secondary loop is assumed to be constant during the transient.

The transient analysed caused by the assumed neutronic perturbation is the increase in the thermal power generated in the core, which reaches up to 260 MW in 5 seconds. This increase in turn causes a cascade increase in fuel temperatures, clad and core outlet temperature. Due to the strongly negative feedback coefficient of the fuel and of the core, the fission power begins to decrease despite the fact that the reactivity insertion is still in progress. It appears that the qualitative trend of fission power, fuel and clad temperatures is the same. It is also noted that the effect of the external perturbation is completely null in 2000 seconds with the reactor reaching a new power equilibrium point.

Finally, it should be noted that, nevertheless the conservative assumption of this transient, from the results shown above it can be seen that the first two barriers should remain intact i.e., the peak temperatures of the fuel and the clad remain below the respective melting and failing temperatures.

References

[1] Frignani, M., Alemberti, A., Tarantino, M. (2019) ″ALFRED: a revised concept to improve pool related thermal-hydraulics″, *Nucl. Eng. and Design*, Vol. 355, 110359.

[2] Alemberti, A., Caramello, M., Frignani, M., Grasso, G., Merli, F., Moressi, G., Tarantino, M. (2020) ″ALFRED reactor coolant system design″, *Nucl. Eng. and Design*, Vol. 370, 110884.

[3] Grasso, G., Lodi, F., Romojaro, P., Garcia-Herranz, N., Alvarez-Velarde, F., Lopez, D., Bubelis, E., Bandini, G. (2018) ″Stress-testing the ALFRED design – Part I: Impact of nuclear data uncertainties on Design Extension Conditions transients″, *Progress in Nuclear Energy*, Vol. 106, pp. 372-386, https://doi.org/10.1016/j.pnucene.2018.03.013

[4] European Commission (2022) ″Advanced Nuclear Safety Evaluation of Liquid Metal using Systems (ANSELMUS) project″, https://cordis.europa.eu/project/id/101061185

[5] Grasso, G., Sarotto, M., Lodi, F., Castelluccio, D.M. (2019) ″An improved design for the ALFRED core″, *Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP)*, pp.12-15, France.

[6] Topliss, J.R., Palmer, I.D., Abeta, S., Irisa, Y., Yamate, K. (1997) ″Measurement and analysis of MOX physical properties″, IAEA TECDOC No. 941, pp.205-215, IAEA, https://inis.iaea.org/collection/NCLCollectionStore/_Public/28/047/28047128.pdf

[7] Luzzi, L., Lorenzi, S., Pizzocri, D., Rozzia, D., Aly, A., Del Nevo, A. (2014) ″Modeling and Analysis of Nuclear Fuel Pin behaviour for Innovative Lead Cooled FBR″, ENEA Tech. Report RdS/PAR2013/022,https://iris.enea.it/retrieve/dd11e37c-db23-5d97-e053-d805fe0a6f04/ADPFISS-LP2-054.pdf

[8] Sobolev, V. (2007) ″Preliminary Design Variants for fuel rod, assembly and core of ELSY-600″, draft tech. Report, ELSY Consortium, SCK-CEN.