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PRELIMINARY ASSESSMENT OF THE FLUID-STRUCTURE INTERACTION EFFECTS IN A GEN IV LMR

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ABSTRACT

The devastating Fukushima 2011 earthquake and tsunami and their consequent multi-reactor damages in Japan, mainly due to the hydrodynamic effects arisen from the fluid-structure interaction, had a significant impact on the global nuclear energy industry.

These events highlighted the need to design the future/existing nuclear installations in order to be able to assure a huge safety level in reference accident configuration and, also, in beyond design conditions.

In this framework it is extremely important to analyze the lessons learnt from the Fukushima events and to evaluate the safety margins of the nuclear power plants in particular, under ongoing unexpected severe earthquake, such as a beyond design basis one (BDBE).

The aim of this paper deals with the evaluation of the dynamic effects seismically induced by the fluid-structure interaction in an under development Gen IV Liquid Metal Reactor (LMR), specifically with reference to the European reactor configuration.

The fluid-structure interaction and sloshing phenomena were numerically analyzed taking into account the non linearities and instabilities due to the influence of material and geometrical parameters.

In order to attain the mentioned goal a suitable numerical procedure based on an external coupling between dynamic and structural codes (as MSC.Dytran and MARC) was applied, setting up a detailed 3-D FEM model as well as implementing a specific algorithm capable to analyze the coupling effects between the considered fluid and the structures and the sloshing phenomenon.

Numerical results were presented and discussed highlighting the importance of the fluid-structure interaction effects in terms of stress intensity as well as the capacity of internals and vessel walls to withstand wave's impacts.

INTRODUCTION

The dramatic consequence of the 9 magnitude Fukushima earthquake highlighted and confirmed that the existing and the future nuclear installations should be designed to be highly secure and capable to withstand a wide range of internal and external extreme loads, such as earthquakes, tsunamis, hurricanes, flooding, etc. Furthermore as the recent Fukushima accident showed (Figure 1), exceptional extreme events are not impossible, even if very unlikely, and can seriously impair the safety of the nuclear facilities, if not correctly taken into account in the design phase.

Figure 1 Fukushima earthquake induced effects

Therefore it is extremely important to analyze the lessons learnt from the Fukushima events in order to (re)assess the safety margins of the nuclear power plants against unexpected severe earthquake, known as beyond design basis earthquake (BDBE), (that could threaten the integrity, the tightness and the operability of safety relevant nuclear SSCs).

The aim of this paper deals with the evaluation of the dynamic effects seismically induced by the fluid-structure interaction in an under development Gen IV Liquid Metal Reactor (LMR), specifically with reference to the European reactor configuration (ELSY or ALFRED reactors).

NOMENCLATURE

BDBE	[-]	Beyond design basis earthquake
SFR	[-]	Sodium Fast Reactor
LMR	[-]	Liquid metal reactor
LFR	[-]	Lead cooled Fast Reactor
GFR	[-]	Gas cooled Fast Reactor
GIF	[-]	Generation IV International Forum
ELSY	[-]	European Lead-cooled System
RV	[-]	Reactor Vessel
SG	[-]	Steam generator
PP	[-]	Primary Pump
DHR	[-]	Decay heat exchanger
RB	[-]	Reactor Building
ELFR	[-]	European Lead Fast Reactor

DESCRIPTION OF THE ALFRED LEAD-COOLED FAST REACTOR (LFR)

To attain the objective to develop a more sustainable nuclear technology which will make the use of nuclear energy through more efficient use of uranium resources (with recycling of Plutonium) and by the reduction of the radio toxicity of the ultimate radioactive wastes, three fast neutron Generation IV reactor concepts, namely, the Sodium Fast Reactor (SFR), the Lead cooled Fast Reactor (LFR) and the Gas cooled Fast Reactor (GFR) are being taken into account in Europe.

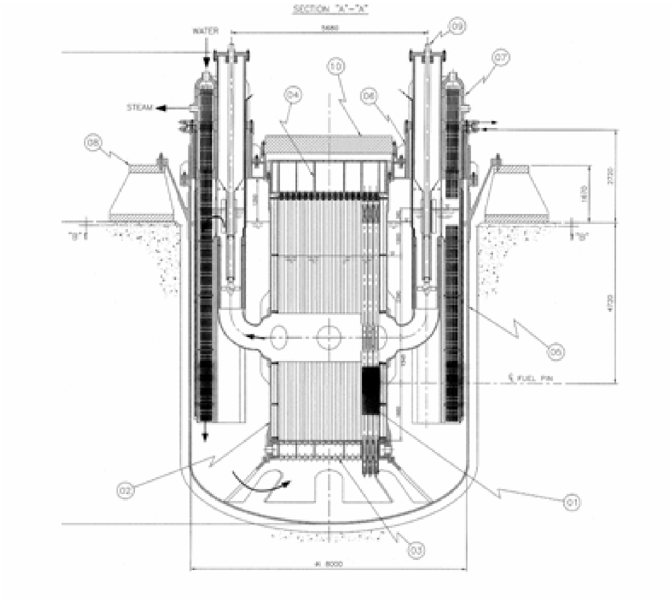
Among the promising reactor technologies being considered by the Generation IV International Forum (GIF), the Lead-cooled Fast Reactor (LFR) has been identified as a system with great potential to meet needs for both remote sites and central power stations. The LFR promises to readily meet the Generation IV objectives of sustainability, economics, safety and reliability, based both on the inherent features of lead as a coolant and on the specific engineered designs [1].

In this study the European ALFRED (lead cooled) reactor configuration, was considered (this work is done in the frame of the LEADER project (EU 7th FP).

The molten lead, as primary coolant, offers good neutronic performance, is chemically inert with air and water and exhibits low vapor pressures with the advantage of allowing operation of the primary system at atmospheric pressure. Moreover lead is compatible with the existing clad material T91.

The ALFRED configuration [1], shown in Figure 2, is based on the results achieved in the ELSY (European Lead-cooled System) project [2-3-4]. The reactor vessel is characterized by an integral shape ("pool type"), housing all the primary system components. The fuel assemblies are supported at their bottom end by a diagrid structure and fixed at their upper end in the cold gas space.

The reactor vessel (RV) is a cylindrical shell with a hemispherical bottom head. The upper part is divided into two branches by a "Y" junction: the conical skirt, that supports the whole weight of RV and its internal components, and the cylindrical one, which only supports the reactor roof. The reactor roof allows in turn to sustain in their correct position and to support all the vessel internals including the steam generators (SGs), the primary pumps (PPs), the core (core barrel and fuel elements) and the decay heat exchangers (DHRs).



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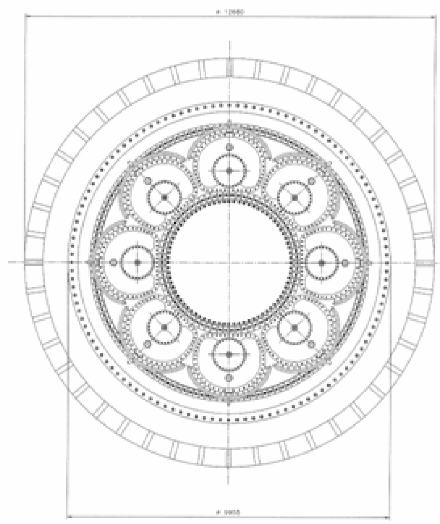


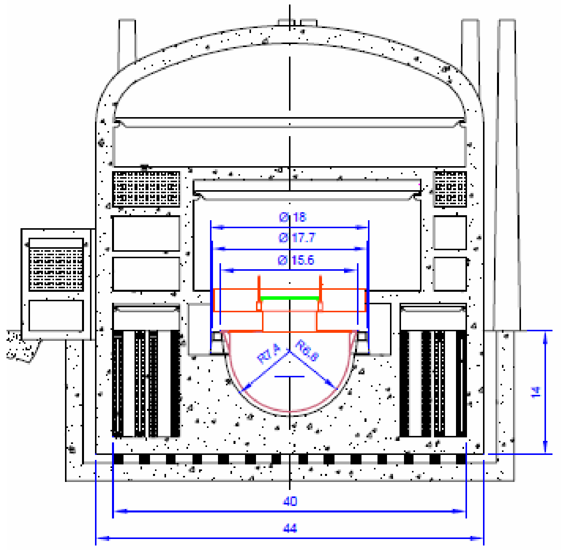
(b)

Figures 2 ALFRED configuration - vertical view (a) and top view (b) [1]

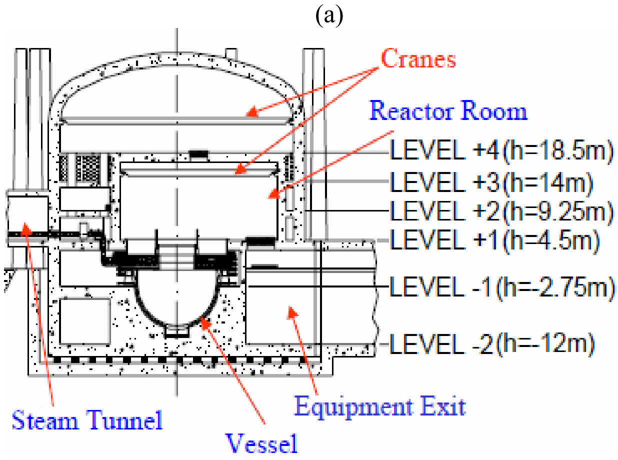
This innovative reactor is moreover characterized by 8 innovative SG (with coaxially pump) systems having a spiral-wound tube bundle arranged at the bottom of the annular space formed by a vertical outer and an inner shells.
The inlet and outlet ends of each tube are connected to the feed water and steam headers, respectively, both installed above the reactor roof .

The reactor building (RB) is assumed to have the same design configuration of that one proposed in the ELSY project (Figure 3), whose main dimensions are:

- External diameter: ∅ = 44 m;
- Height: h = 48.5 m;



(a)



(b)

Figures 3 Reactor Building main dimension and general configuration (b)

The reasons for such a large diameter of the reactor building are due to the large vessel dimensions, about 18 m (related of course to the pool type integral configuration), to the reactor room, the decay heat removal system pools, etc.

Moreover the reactor building has been considered to be fixed at ground level.

The use of a compact solution for the RV and a simplified and innovative primary circuit, characterized by the possibility to remove all the internals, are useful to mitigate the possibly adverse effect of the high density of lead [5].

The primary system design temperature is 400°C and the design pressure about 1 bar. The secondary side operational condition range of the SG tubes is between 335°C and 450°C at about 20 MPa, while the primary coolant temperature is 480°C at the core outlet.

The reactor vessel, the skirt and the SG outlet are made of SA 240 316LN, while the SG support box and base plate are made of SA 516 Gr 70 carbon steel.

Important key parameters of ALFRED reactor are summarized in the following Table 1.

Table 1- key parameters of ALFRED

Power	300 MWth (~120 MWe)
Thermal efficiency	40% (or better)
Primary coolant	Pure lead
Primary system	Pool type, compact
Primary coolant circulation	Forced (mechanical pumps)
Primary system pressure loss	< 1.5 bar
Primary coolant circulation for DHR	Natural circulation
Steam Generators	8, integrated in the main vessel
Secondary cycle	Water-superheated steam at 180 bar, 335-450°C
Primary pumps	8, mechanical, integrated in the SGs, suction from hot collector
Internals	All internals removable
Inner vessel	Cylindrical
Hot collector	Small-volume, enclosed by the Inner Vessel
Decay Heat Removal	2 independent, redundant and diverse DHR systems, 3 out of 4 loops of each system are capable of removing the decay heat
Seismic design	2D isolators supporting the RB

DEVELOPMENT OF BDBE ANALYSIS

The intent of this paper is to provide some contributions to the development of European Lead Fast Reactor (ELFR) configuration (~600 MWe) to be used as a reference for ALFRED project (developed within the EU 7th Framework Project) that will constitute the reference system for the large lead-cooled reactor of Gen IV.

Heavy metal primary coolant, that characterize some NPP type responds to dynamic motions, particularly to the seismic one, and when the excitation has a frequency near the natural one of the container system, rather "violent" waves can form and impact into the tank walls. In particular the impact of waves (hydrodynamic pressure and impact force) on the RV walls and on its internal structure could result in a serious concern, from a structural point of view, because of the high density of lead.

When interacting with its retaining structure, the free liquid surface can exhibit several types of motion in the form of modulated free surface waves and energy exchange between interacting modes.

The analysis of the liquid sloshing and fluid structure interaction are of meaningful importance because of the need to evaluate the safety margin of the reactor structures, systems and components.

The FSI induced by an earthquake event shall be evaluated also in the case the reactor building foundation was provided of efficient seismic isolation devices in order to mitigate the propagation of the seismic dynamic loadings.

As for the structural issues, the fluid-structure interaction problem is investigated in relation to the choice of lead material as primary coolant; the seismic inertia mass of the reactor coolant might significantly increase and result in a severe hydrodynamic pressure acting on the reactor vessel walls.

The dynamic loadings, as already mentioned, may cause unacceptable consequences on the reactor systems, such as the buckling of the RV or of its internals, the over stressing of the roof, in the case of lead wave impact etc.

The basic problem of liquid sloshing involves the estimation of hydrodynamic pressure distribution, forces, moments and natural frequencies of the free-liquid surface. These parameters have a direct effect on the performance (structural integrity) of the considered reactor structures systems and components.

It is worthy to note that, generally, the hydrodynamic pressure of liquids in moving rigid containers has two distinct components:

1) the first component is directly proportional to the acceleration of the tank, since it is caused by the part of the fluid moving with the same tank velocity;

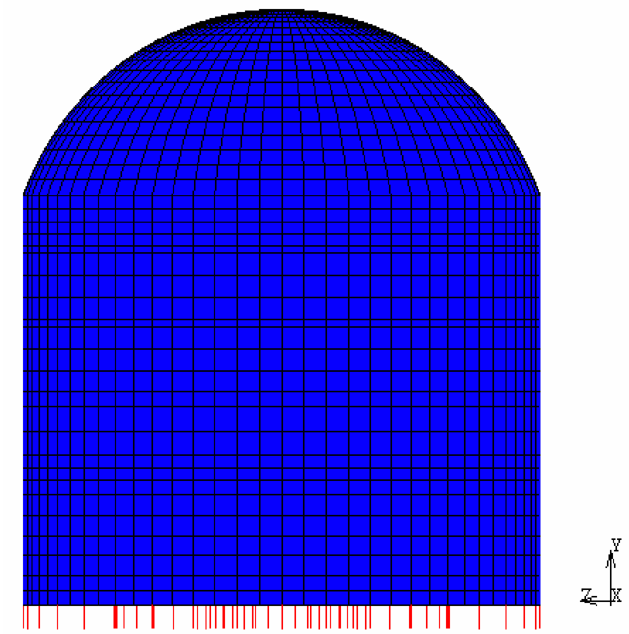
2) the second one known as ‘‘convective’’ pressure that represents the free-surface-liquid motion [6].

Besides it must be pointed out that a realistic prediction of seismic related sloshing phenomenon is made particularly difficult by its non linear nature, characterized by a large number of parameters affecting it, such as the complex reactor geometry, the liquid variable height (during motion), the material properties (not elastic behaviour), etc.

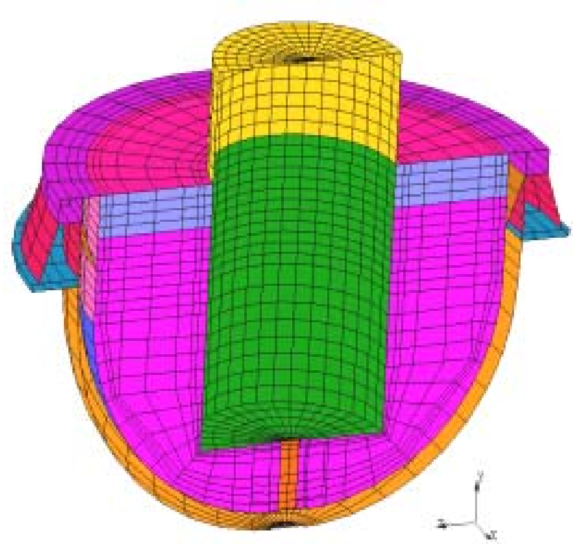
In consideration of these weak points/difficulties, the design philosophy adopted for the evaluation of the seismic capability of the considered LMR is a deterministic approach, based on a numerical evaluation (non linear analysis), by means of finite element method, capable to simulate and evaluate the effects induced by the propagation of the seismic waves on the mainly relevant structure (in terms of seismic demand parameter) and to represent adequately the fluid-structure interaction (FSI) and sloshing phenomena.

NUMERICAL APPROACH

To evaluate the structural performance of ALFRED reactor subjected to a BDBE, a conservative analysis has been carried out because only the cylindrical inner vessel (“core region”) has been considered; therefore the lead mass inventory was over-estimated. To the intent the Time History and the Substructure approaches were applied. Rather complex models (Figures 4a and b) representative of the RB and of the main and mutually interacting reactor components were set up and implemented.



(a)



(b)

Figures 4 Reactor Building (a) and RV (b) preliminary models

The considered and modelled structures, systems and components, previously shown in Figures 4a and b, are the followings:

- the Reactor Building;
- the Safety Vessel with its annular box structure;

- the Reactor Vessel and its support system;
- the molten primary coolant: pure lead;
- the cover gas: argon.

In the present study the T91 martensitic steel, also pre-selected for the design of EFIT and XT-ADS European facilities, has been considered.

To correctly represent the behaviour of the considered LFR reactor the experimental mechanical properties of T91 steel, as shown in Figure 5, were assumed as input in the implemented model [7]. In Fig. 5 it is, indeed, represented the "*..engineering stress-strain curves obtained with T91 specimens, after standard heat treatment, after 4000 h pre-exposure to LBE at 450°C…*" as quoted in NEA handbook [7].

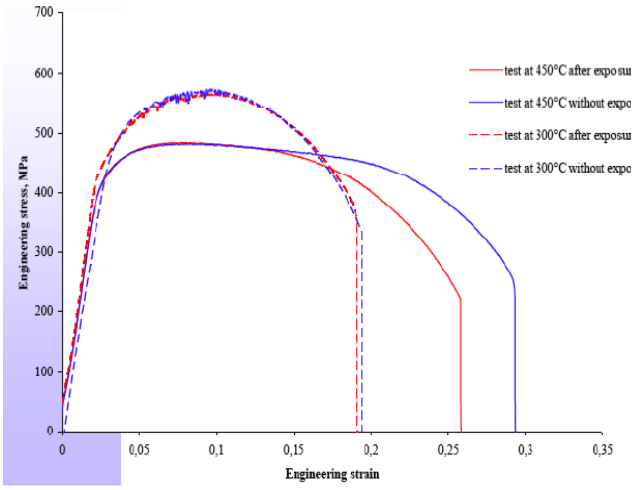


Figure 5 Experimental stress-strain curves for T91 [7]

In this paper seismic related fluid-structure interaction problem was investigated by means of an appropriate dynamic finite element code (MSC.Dytran© [8]) implementing the Lagrangean-Eulerian Algorithm (ALE) which allowed to solve the equations of the fluid motion [9] at each point and time step.
In Figure 4b it is represented the implemented 3-D finite element model used to simulate the behaviour of the main reactor structures, undergoing the seismic excitation, and the sloshing effects, according to the mentioned ALE approach. Moreover Eulerian hexahedron elements were chosen to implement the primary coolant and the cover gas, while the reactor vessel and internals structures by means of Lagrangean shell elements.

BDBE transient analysis

The preliminary analysis was carried out in two step: the first one allowed to evaluate the influence of the dynamic loads propagating through the isolated reactor building; the second one allowed to analyze the structural effects induced by the ground motion on the RV and its main internal components.
The isolation may be obtained using an iso-elastic approach (isolators were represented by means of springs coupled to dashpots capable to simulate the behaviour of high damping rubber bearing components), and assuming an isolator's frequency equal to 0.5 Hz.

In order to understand the dynamic response of the building and to evaluate its dynamic characteristics the same input Acceleration Time Histories (ATHs) were applied at the base of the foundation of the isolated RB. The input acceleration data were elaborated according to the updated Regulatory Guide US NRC 1.60 and 1.92 [10], considering a 5% of critical damping value. They were represented in Figure 6 by means of three artificial time histories components, two along the horizontal direction (Ax and Az) and one along the vertical one (Avert), compatible with the given free-field spectra which represent the assumed BDBE at a hypothetical embedment in stiff rock.
The vertical acceleration (indicated as Ay in Fig. 6) was also assumed equal to 2/3 of the horizontal one in the entire frequency range.

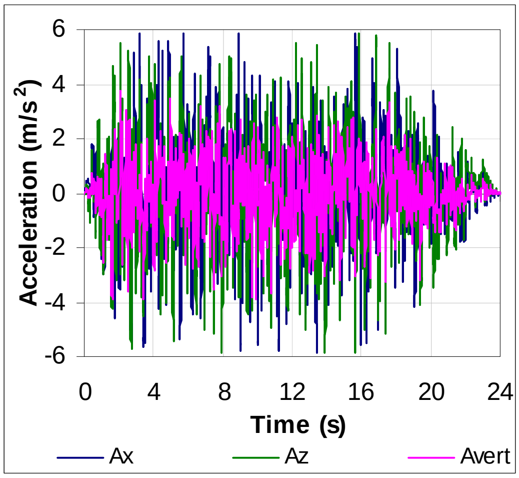


Figure 6 Input Acceleration Time Histories

BDBE analysis results and discussion

Before the evaluation of the effects due to the dynamic forces exerted/induced by the fluid motion coupled to the FSI on to the RV structures, the influence of the dynamic loads propagating through the isolated reactor building was carried out.

Preliminarily a modal analysis was performed to check the consistency between the isolated RB structure and the isolation system and confirm that the considered RB structure behave as a "rigid body". Subsequently suitable seismic transient non linear analyses were carried out in order to calculate the acceleration values propagated up to the anchorage of the safety vessel.

Overviews of the obtained acceleration values allowed also to confirm the favorable effects of the isolation system in mitigating the propagation of the accelerations inside and along the RB containment structure (Figure 7): reduction of about 40-50%.

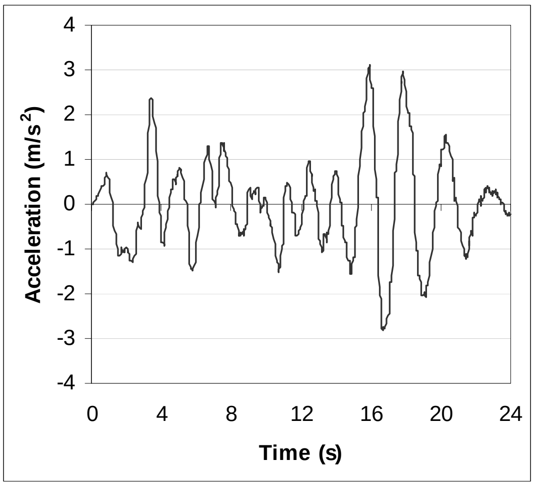


Figure 7 Horizontal acceleration at the SV anchorage

It is important to highlight that due to the reactor vessel height (to be considered like an "elevated structure"), the large mass and free surface of the lead, the sloshing phenomenon may become very important because it might produce stresses exceeding the allowable limits in localized parts of the reactor internals components and, therefore, impair their integrity.

In the performed analyses, only the transmitted horizontal acceleration was used as input in the RV substructure (previous shown in Figure 4b) in order to analyze the structural effects induced by the ground motion on the RV and its main internal components, taking into account the effects of the moving lead. The main assumptions made are:

- Fluid has an elastic, linear, isotropic behaviour;
- Lead is modelled as Eulerian fluid
- RV, SV and internal structures behaviour was linear elastic perfectly plastic as well as isotropic;
- Fluid and structure may exchange mechanical energy at the fluid-structure interface;
- The fluid-structure coupling is treated using the Arbitrary Lagrangean Eulerian;
- Argon is modelled as an ideal gas;
- BDBE input motion is represented through the horizontal velocity corresponding to the SV ATH (along the x axis direction), because of the feature of Dytran© code [8].

The coupling effects between the fluid and the surrounding structures was calculated by means of the Arbitrary Lagrangean Eulerian coupling algorithm. This algorithm allows to define an interface surface, that also serves as a boundary for the flowing Eulerian material during the analysis.

Moreover, as already mentioned the carried out simulations may be considered rather conservative because in the performed analyses the RV model did not include all internals structures and components, therefore the obtained results refer to a more conservative evaluation of the fluid-structure interaction between the reactor vessel and lead coolant and sloshing effects.

The preliminary results (structural effects and consequences) obtained from the carried out seismic analyses, are presented in the following figures and discussed in order to highlight the importance of the fluid-structure interaction phenomenon in terms of stress intensity distribution inside the RV and internal components as well as of the fluid movement along/inside the vessel (due to the impulsive and convective-sloshing components of the fluid motion).

It was observed that the elevation of waves, about 10 cm was not sufficient to impact the roof.

Lead motion coupled to the propagation of seismic wave resulted in a stress intensity distribution that could impair the structures capability to withstand the related dynamic loads on the RV and internal components. Moreover it was observed that the inner cylindrical vessel (which allows to enclose and sustain the core) structure seemed to influence the fluid waves motion by fragmenting the fluid wave.

The fragmentation allowed also to avoid that a more extensive lead mass could impact the roof: subsequently the impact force is reduced as well as the risk of structural damage. Another aspect that determined a further reduction of the impact force is the drug of the argon gas into the lead during the fluid motion due to the resulted variation of lead density (at 6 s, as an example), clearly visible in Figure 8 around the yellow interface.

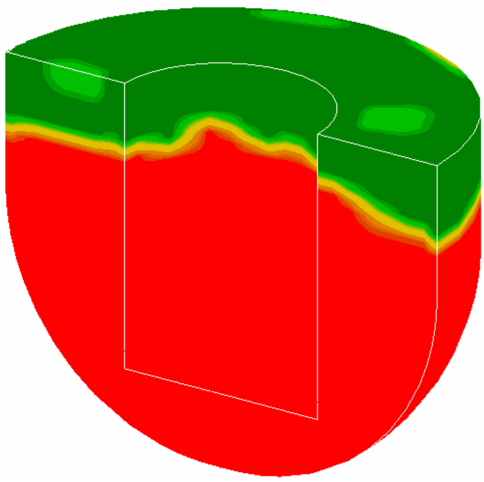


Figure 8 Lead density variation behavior

Figure 9 shows the hydrodynamic pressure distribution into the reactor vessel due to the lead motion; it highlights that the mean pressure values range from about 1 to 2.5 MPa: this variation seemed to depend on the level of seismic motion intensity.

Moreover the maximum pressure value (≈ 6 MPa at t ≈ 4 s) occurred on the bottom of the reactor vessel and of the inner vessel. Although this high value, the seismic buckling of the reactor vessel and its internals is prevented, for the reason that the seismic pressure greatly increases as the coolant depth becomes deeper.

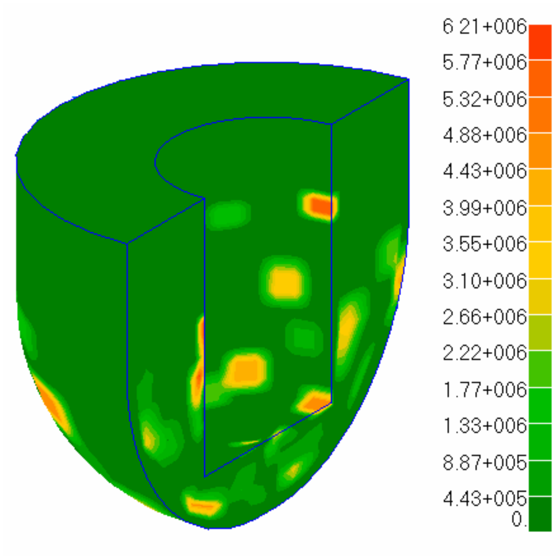


Figure 9 Pressure distribution inside the RV at t ≈ 4 s

The progressive lead motion and in particular the formation and impact of lead waves (hydrodynamic pressure and the fluid movement characteristics) seemed to determine high Von Mises stress values (Figure 10) in the reactor vessel and its internals walls. The maximum stress values resulted about 210 MPa and localized in correspondence of the inner cylindrical vessel walls.

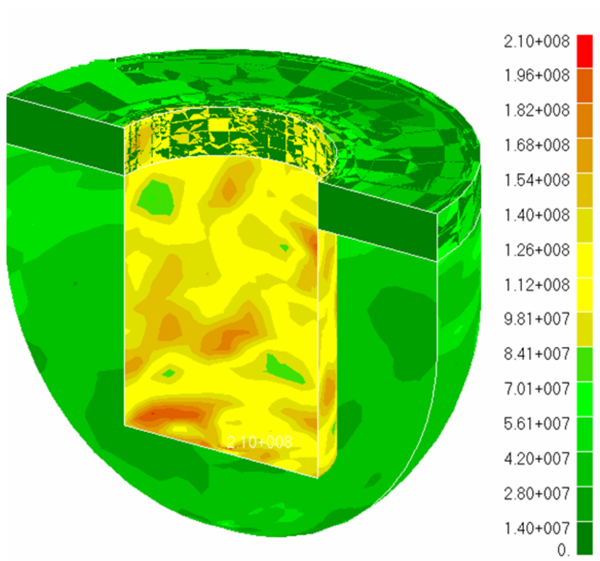


Figure 10 Von Mises stress distribution inside RV

In Figure 11 it is represented the calculated and smoothed behaviour of Von Mises stress; this latter is much more important from a structural point of view because does not contain the vibration component.

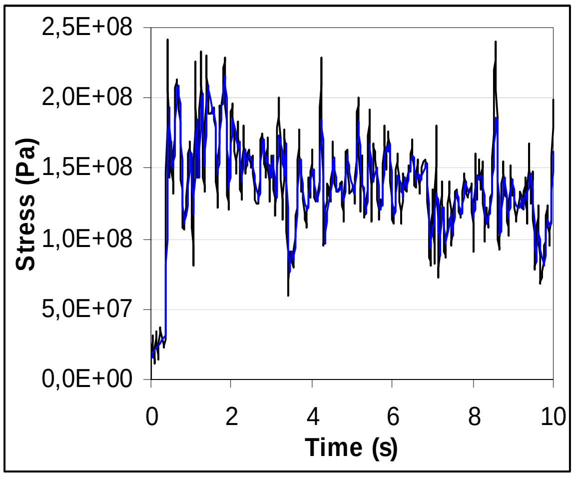


Figure 11 Von Mises stress behaviour at the inner vessel wall

It is important also to note that the stress values calculated at the inner vessel wall, probably induced by the fluid movement and/or fluid wave impact on the RV structures, were anyway not sufficient to determine the plasticization of the RV and inner vessel wall thickness and therefore to impair their structural integrity in view of the ASME code rules.
In addition it is important to consider that in the performed analyses the fluid is assumed to fill a rather more extensive region inside the vessel therefore the obtained stresses might be greater than the real ones.

CONCLUSION

In this report the results of preliminary seismic analyses are discussed, as obtained using the Time History method coupled to the substructure approach that allowed to study separately a hypothetical ALFRED containment building and the reactor vessel with the inner cylindrical vessel.
To perform the analyses, appropriate Substructure approach with 3-D FEM models, representative of the isolated reactor building and of the safety and reactor vessels, etc., were set up in order to evaluate the seismic response of the structures and internal components that are particularly sensitive to the seismic events due to the large coolant mass in LMFR.
In the carried out preliminary analyses, the effects of the coupling between the fluid and the reactor vessel structure both in terms of the stresses level and distribution were presented.
The input acceleration may determine the arise of fluid sloshing waves that may induce relevant hydrodynamic pressures on the RV and internal components walls which, in turn, generate a corresponding stress intensity distribution.
The obtained numerical results, for implemented models, highlighted that:

1) the maximum Von Mises stress values seem to be located at the bottom of the inner cylindrical vessel;
2) the obtained RV internal displacements, due to the deformation induced by the fluid motion, are rather large and highlight a criticality in the reactor internals design,

while the displacement of the SV and RV ones are negligible;

3) The sloshing analyses performed up to now have highlighted the need to improve the structural design of primary system components, however with no significant modification of their functional geometry or layout within the main vessel;
4) The fluid-structure interaction effects have been thus proved of meaningful importance in the dynamic behaviour of the reactor pressure vessel with heavy coolant fluid.

The set up model, even if used to simulate the fluid-structure interaction, includes some relevant internal components; nevertheless it may be useful to further upgrade the reactor vessel and internal design.

REFERENCES

[1] Mansani L.,"System Configuration for LFR", *International Workshop on LMFRs*, Aix-En Provence, October 4-7, 2011.

[2] Cinotti L., Smith C. F., Sienicki J. J., Ait Abderrahim H., et al., "The potential of the LFR and the ELSY Project", RGN, n° 4, Juillet-Aout 2007.

[3] Cinotti L., Smith Craig F., Sekimoto H., "Lead-cooled Fast Reactor (LFR) Overview and perspectives", *Proceedings of the GIF Symposium 2009*, Paris, France Sept. 9-10, 2009.

[4] Alemberti A., Carlsson J., et. al. "European lead fast reactor – ELSY", *Nuclear engineering and design*, vol. 241, 2011, 3470-3480.

[5] Mourogov V., Juhn P. E., Kupitz J., Rineiskii A. "Liquid-metal-cooled-fast reactor (LMFR) development and IAEA activities". *Energy*, vol. 23, Issues 7-8, 1998, 637-648.

[6] Ibrahim Raouf A, "Liquid Sloshing Dynamics", *University Press* Cambridge, 2005.

[7] NEA\AEN, "Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermal-hydraulics and Technologies", *OECD Publications*, 2007.

[8] Dytran© Software, User's guide, 2010.

[9] Hernandez-Barrios H. et al., "Nonlinear sloshing response of cylindrical tanks subjected to earthquake ground motion"; *Engineering Structures*, 29, 2007, 3364-3376.

[10] U.S. Nuclear Regulatory Commission, , "Design Response Spectra for Seismic Design of Nuclear Power Plants", *R. G. 1.60, Rev. 1*, Washington, D.C., December 1973.