Agenzia nazionale per le nuove tecnologie, l'energia e lo sviluppo economico sostenibile

MINISTERO DELLO SVILUPPO ECONOMICO


# Messa a punto e sviluppo di metodologie e analisi per la valutazione preliminare del fenomeno della "core compaction"

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CIRTEN - CONSORZIO INTERUNIVERSITARIO PER LA RICERCA TECNOLOGICA NUCLEARE

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MESSA A PUNTO E SVILUPPO DI METODOLOGIE E ANALISI PER LA VALUTAZIONE PRELIMINARE DEL FENOMENO DELLA "CORE COMPACTION"

G. Forasassi, R. Lo Frano, G. Pugliese (UNIPI)

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Il presente documento descrive le attività di ricerca svolte all'interno dell'Accordo di collaborazione "Sviluppo competenze scientifiche nel campo della sicurezza nucleare e collaborazione ai programmi internazionali per il nucleare di IV generazione"
Responsabile scientifico ENEA: Mariano Tarantino
Responsabile scientifico CIRTEN: Giuseppe Forasassi

**Titolo**

# Messa a punto e sviluppo di metodologie e analisi per la valutazione preliminare del fenomeno della "core compaction"

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**Descrittori**

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**Sommario**

Nuclear reactors in operation have to be maintained in a critical state in order to keep the chain fission process stationary and under control. The safety priority is therefore to keep the reactivity known and under control. Nuclear stability considerations dictate that the geometry of the core be closely controlled at all times. This study focuses on the deformations of core (and restraint system) geometry due to the dynamic perturbations. Since the primary function of the core is to provide the reactivity under control, any modification of its geometry must be predictable and safe. The same constraint must be fulfilled by the core restraint system that must be/remain compatible with the requirements of the interfacing reactor systems. It is therefore essential to set up/develop an overall and reliable methodological approach to be used in designing the core system (all structures and components characterizing the core region) and evaluating its performance under operation and accident condition. Modelling the dynamic behaviour of a LMR core is particularly needed for seismic design purpose and more generally for the study of dynamic solicitations due to internal or external accidents. A special attention should be given to these solicitations that could deform the behaviour of core system and of each fuel assembly.This preliminary study firstly presents a way to simplify the problem, by adopting substructure approach to preliminary and general analyse the dynamic behaviour of a LFR core region. To the aim a simplified numerical modelling was adopted, in which the fuel elements and assemblies were represented as lumped mass distributed on the fuel supporting plate; contact points between adjacent assemblies as well as the thin fluid layers between assemblies were not considered at this stage of the assessment. The obtained preliminary numerical results, for the implemented models, highlighted that the displacements at the core region resulted of about 2 cm. Sensitivity analysis was also carried out to evaluate the influence of the mesh size and element type on the dynamic response of structures analysed. Future further developments are of course necessary to determine with more accuracy the deformations of (sub)assemblies and element/pad cross-section undergoing compaction.

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# Core compaction phenomenon: methodological approach

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# Summary

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This study focuses on the deformations of core (and restraint system) geometry due to the dynamic perturbations. Since the primary function of the core is to provide the reactivity under control, any modification of its geometry must be predictable and safe. The same constraint must be fulfilled by the core restraint system that must be/remain compatible with the requirements of the interfacing reactor systems.
It is therefore essential to set up/develop an overall and reliable methodological approach to be used in designing the core system (all structures and components characterizing the core region) and evaluating its performance under operation and accident condition.
Modelling the dynamic behaviour of a LMR core is particularly needed for seismic design purpose and more generally for the study of dynamic solicitations due to internal or external accidents. A special attention should be given to these solicitations that could deform the behaviour of core system and of each fuel assembly.
This preliminary study firstly presents a way to simplify the problem, by adopting substructure approach to preliminary and general analyse the dynamic behaviour of a LFR core region.
To the aim a simplified numerical modelling was adopted, in which the fuel elements and assemblies were represented as lumped mass distributed on the fuel supporting plate; contact points between adjacent assemblies as well as the thin fluid layers between assemblies were not considered at this stage of the assessment.
The obtained preliminary numerical results, for the implemented models, highlighted that the displacements at the core region resulted of about 2 cm.
Sensitivity analysis was also carried out to evaluate the influence of the mesh size and element type on the dynamic response of structures analysed.
Future further developments are of course necessary to determine with more accuracy the deformations of (sub)assemblies and element/pad cross-section undergoing compaction.

# 1. Introduction

One of the objectives of advanced liquid metal reactors (LMR) is to maximize the safety of reactors in any design and beyond design condition for all plants lifetime, so LMR designs are expected to significantly increase the safety [1] [2].
It is important to understand the deformation behaviours of LFR core mainly in the aspects of safety, i.e. the reactivity change in the transient period, the fuel life prediction, etc., in order to attain more precise information for sophistication of the core structure design.
To secure the control rod insertion, the core structural integrity, and an accurate prediction of the reactivity insertion, the core seismic analysis should be carried out with a highly accurate method which can take into account the non-linear behaviour, especially related to the fluid-structure interaction.
The safety priority is therefore to keep the reactivity known and under control. Several sources of non controlled reactivity perturbations may be:

a) the quasi static perturbations (having large time constant ranging from 1h to 1y) the reactivity impact of which may be compensated by the control rods;

b) the dynamic perturbations having small time constant (1s),which require a fast response from the safety system of the core, by compensation or emergency stop.

The compaction of core assemblies is a crucial aspect of the core system design because of the consequences and effects (as an example the loss of flow, etc.) on the net reactivity, especially, during transient conditions caused by external events.
As a result of an external event condition (dynamic loading propagated to the core) related to an earthquake or flooding, vibration induced by an airplane crash, etc., the core assemblies may distort, along the height in the radial and circumferential directions, relative to their nominal positions.
This deformation may determine, at large or small extent, an assembly compaction, which is generally characterized by a radial inward displacement (flexion + constraint on pads), and, subsequently, results in a possible insertion of reactivity. For this reason the core compaction is considered a reference design accident event. Moreover the core compaction phenomenon should occur in a very short time period, of about 1 s as indicated at the previous point b), even if, in reality, the duration and evolution of this accident scenario is unknown.
Therefore, it is notably necessary to ensure that the effects of dynamic loadings on the core can be adequately managed [2]:

- In any situation, a large compaction of the fissile material must be avoided. Therefore when subjected to any kind of dynamic loading relative movements between fissile subassemblies and absorber rods should be limited, and fuel pins break prevented. Moreover, it shall be demonstrated that the reactor must safely shut down: the control rods must be able to fall within the core during a dynamic excitation.

- Core cooling capacity of the assemblies in the core region must be demonstrated under dynamic loading.

- Material containment is assured, particularly during earthquake it must be demonstrated that fuel pins (first barrier to prevent radioactive release to the environment) are not damaged.

To deal with these issues, the designer must rely on numerical models of the core dynamic behaviour.
Reactor safety and core performance considerations, relative to assembly positions, internals design features and interactions, find expression in the definition of core restraint requirements.

## 1.1. Physical problems to be addressed

In the design of fast reactor cores, the positions of core assemblies and their interactions with each other and with the surrounding internals structure must be known and controlled in order to assure adequate safety and core performance.
This concern gives rise to the definition of core restraint requirements which form the basis for the selection of specific core restraint design features in core and internals components [3][4].
The major core restraint requirements include the following:

**a. Reactivity**

Small lateral motions of the core region of fuel and radial blanket assemblies, due to thermally induced bowing, compaction and expansion, produce nuclear reactivity variations which can significantly affect reactor transient response.
To enhance reactor stability and inherent core safety, core assemblies have to be supported such that "motions" are possible due to the existence of finite inter-assembly and peripheral gaps.
Sudden core motions due to seismic excitation have the potential of introducing pulsating or step reactivity insertions into the core. The gap structure of the core must therefore be limited to assure that resulting reactor power excursions do not cause fuel design limits to be exceeded.

**b. Assembly Deformation**

In the nuclear and thermal environment of the LMFBR core, structural materials are subject to swelling and creep induced as irradiation effects. The high temperature and neutron flux environment together with lateral thermal and neutron flux gradients in the core region produce duct dilation and permanent assembly bowing.
The deformation (Figure 1) may be otherwise induced by dynamic or vibration loads resulting in an inward/outward assembly deformation (that could be localized and/or partially).
These effects can impact the capability to assure and maintain structural integrity of core components during normal core operation as well as during insertion and withdrawal processes.
Core component design and restraint feature must therefore reflect the need to accommodate such a type of deformation.

**c. Assembly Alignment**

Thermo-mechanical loads can also modify the alignment of the core assemblies with interfacing control and refuelling system components. Therefore at the design stage features should be included in order to assure the alignment of the assemblies within the envelope specified by the interfacing systems and to avoid reactivity insertion.

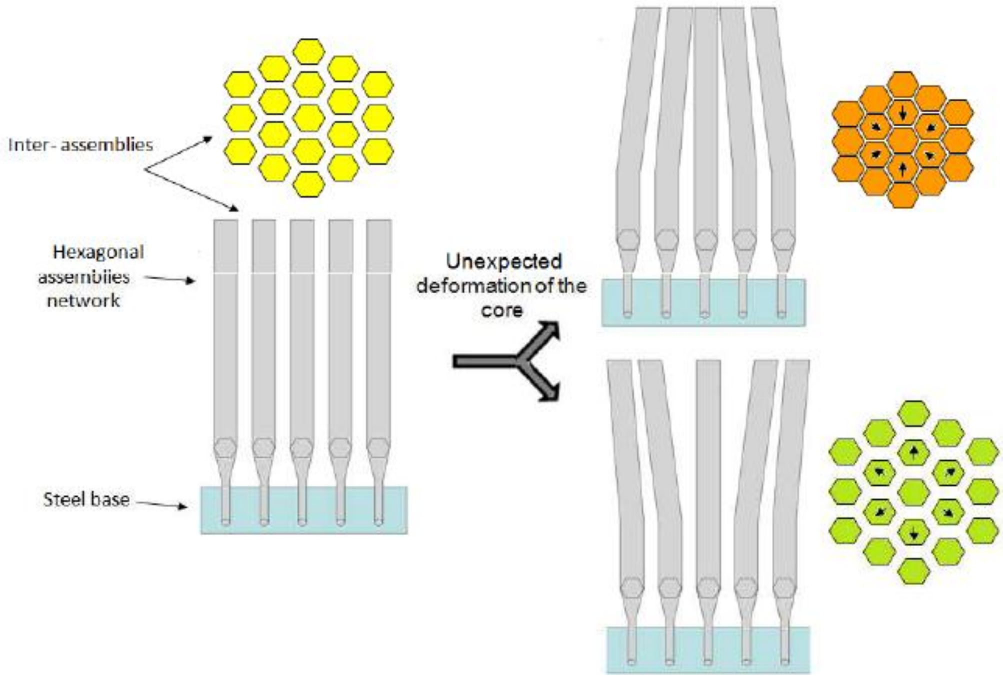



Figure 1 – Scheme of core compaction and flowering.

### d. Load Transmission

The need to limit and control core motions typically results in the definition of load transmission planes. These planes are identifiable at core and internals component elevations at which interfacing gaps are reduced.
At these planes the components must be capable of sustaining and transmitting loads arising from normal reactor operation and accident condition, like an example an earthquake event.
Satisfaction of the core restraint requirements entails the identification and implementation of a suitable set of core restraint features into the component designs.

# 2. SMA formulation

In the formulation of a program for safety evaluation and design support, deterministic methodology plays an important role because this type of approach will allow to evaluate the inherent capability of nuclear plants to resist also severe internal external events, not considered in their original design: by simulating the behaviour of structures, systems and components, in detail and as real as possible in order to determine the strength capacity of the analysed structures.

Conservatisms are also present in such a type of approach due to the fact that it is compounded through the safety analysis and the design chain.

Traditionally, to measure the damages sustained during the dynamic response of a plant, the structural effects of its safety relevant SSCs, caused by accidental events, were determined and calculated in terms of the maximum displacement (or through the maximum ductility factor) as well as in terms of stresses. Therefore the inherent capability or robustness, usually indicated [5] as ‘design margin’ to failure, is a direct measure of the load (structural) bearing capability to be used in the design and qualification procedures of nuclear plant, according to current practices.

It is known the design margin exists and is ensured through the use of design criteria in industry standards and guidelines, particularly those applicable to nuclear installations. Moreover, the evaluation and quantification of the structural capacity (by thermal hydraulics, neutronic, thermo-mechanical and severe accident codes) of an existing or next/under development nuclear installation represents a way to understand the true state of the SSCs in terms of required safety functions and capacities.

The SMA methodology considers, often, a higher level of hazard (high intensity/magnitude of the initiating event) to be conservative and determine/associate it the really strength capacity. In so doing, the inherent additional capacity of the SSCs may be taken into account as well all significant information concerning material property (taking into account failure mechanism and deterioration process), SSCs geometry (progressive damage of component and structure) and operational status, boundary and initial conditions and deriving from all relevant sources, such as vulnerability of selected structures and/or non-structural elements.

In this light, a measure of the damage is mainly associated with nonlinear behavior of facilities which is dependent on many factors, among which the maximum displacement, the inherent straining of the materials at critical sections or elements, the stress level, etc..

The two linear and nonlinear approaches adopted to evaluate the SMA, in view of the imposed initial and boundary conditions, are not necessarily consistent: observations of the physical behavior of structures and structural elements, in the field and laboratory, indicated that the deformation process is difficult to describe by linear approach so nonlinear behavior might be preferred to determine damaging process.

To reduce the uncertainties related to the evaluation methodology, it is necessary to correctly characterize their behavior and determine the factors which mainly may influence it.

The starting point in the definition and setting up of a methodology capable to determine the safety margin (that simply means if the stress level is acceptable or exceeds the allowable limit): the first step is therefore represented by the collection of the general documentation of plant and its SSCs (BOP, lay-out, drawings, site characteristics).

Emphasis should be put on any specific data and information used/to be used at the design stage:

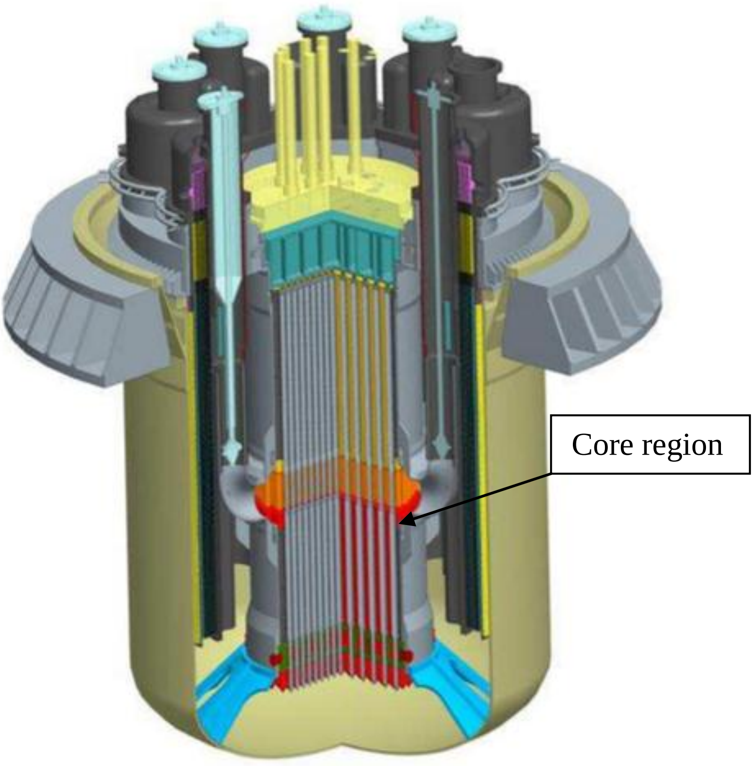
- nominal properties of materials and their mechanical characteristics;

- load combinations, in view of the verification of components;
- design parameters of structures, components, piping systems and other items: general arrangement and layout drawings, typical details, like connections, reinforcement type, leak tightness system, etc.;

The accuracy of the collected input data in order to guarantee lesser uncertainty affect deterministic analysis and to carry out the real behaviour of component analyzed.
Specifically, data or information of the SSCs allowing a realistic simulation of component under transient conditions (dynamic, thermal, TH, etc. one) should take into account system, geotechnical and structural designs.

## 2.1. System design

Figure 3 illustrates the configuration and location of major components in the ALFRED reactor. The core assemblies are seen to be positioned axially and supported by the lower internals structure (mainly by the diagrid).
Lateral support is provided at two elevations by the core barrel. Furthermore an upper restraints allow to maintain the vertical position and guarantee also the safety function by means of rod control system positioning and a tungsten ballast (used for safety shut down).

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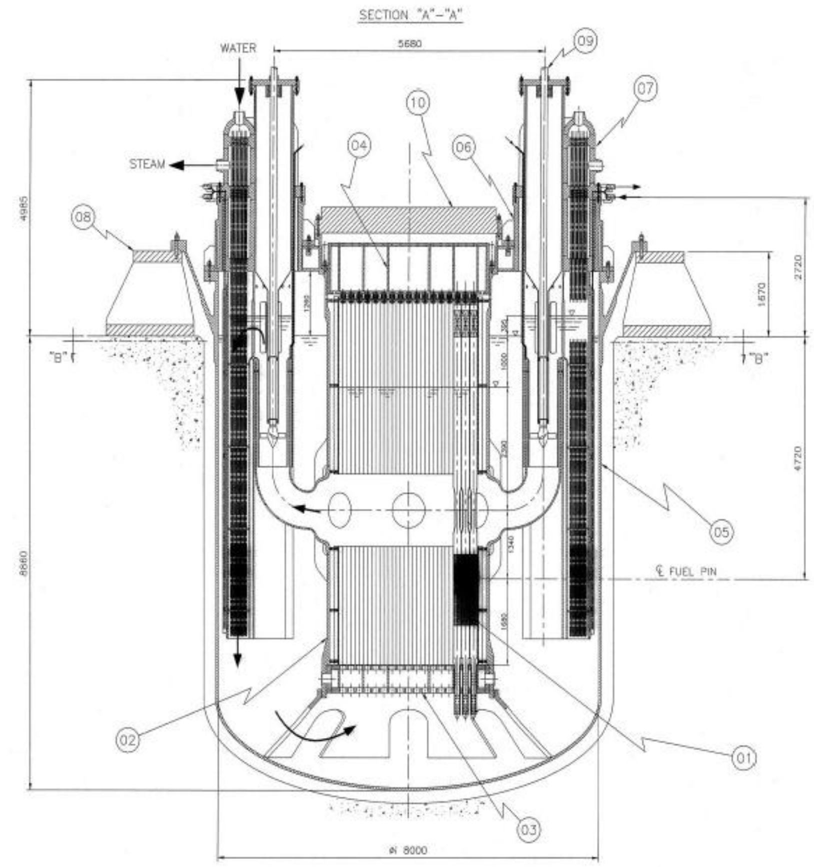



(b)

Figure 2 – ALFRED reactor scheme (a) and elevation view (b).

The reactor building (RB) layout considered in this assessment was the same proposed for the ELSY project [6]. This choice was motivated by the fact that the design of Alfred has not to date involved the part relating to the building of the reactor.
After clarifying this aspect, in the following this reactor building model was briefly described. It has been assumed the RB (Figure 3) main dimensions are:

- External diameter: ∅= 44 m;
- Height: h = 48.5 m;
- Thickness of walls: t ~ 1 m.

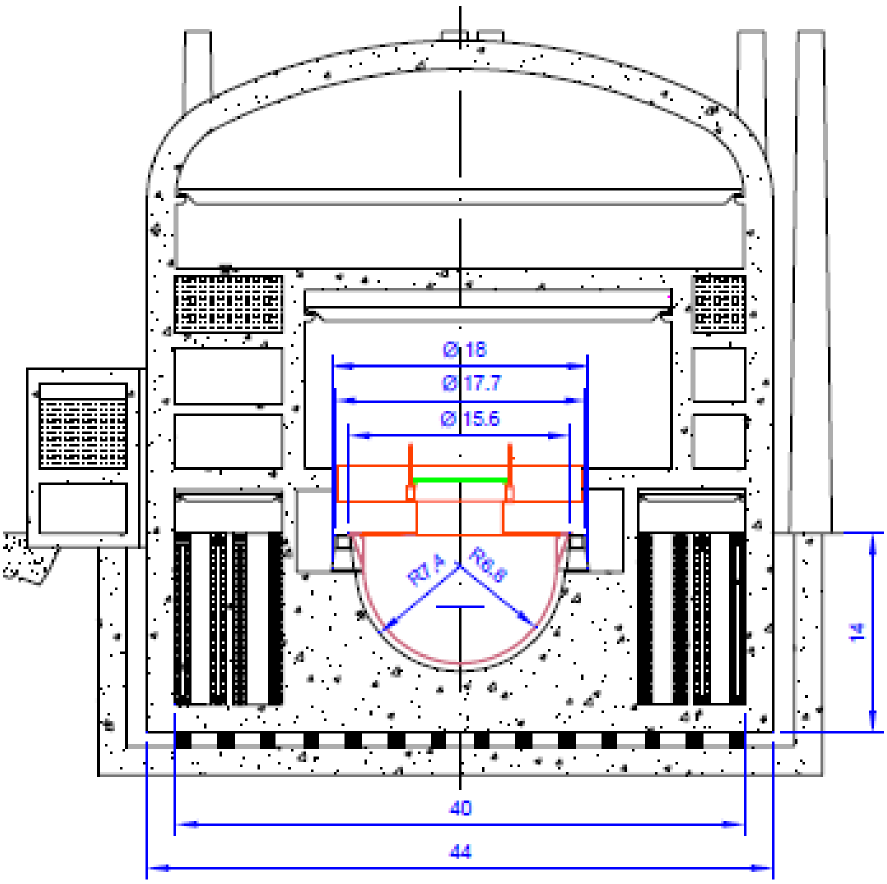
The reasons for such a large diameter of the reactor building are due to the large vessel dimensions, about 18 m (related of course to the pool type integral configuration), to the reactor room, the decay heat removal system pools, etc. Moreover the reactor building was considered to be fixed at ground level and isolated.
The base isolation (BI) strategy adopted to protect the internal structures as well non-structural components allows to guarantee the passive protection of critical facilities under moderate-to-major earthquake ground motions. The BI, obtained by inserting isolators between the foundation of RB and the soil, relies upon the following properties:

- Horizontal flexibility to increase structural period and reduce the transfer of seismic energy to the superstructure (except for very soft sites);

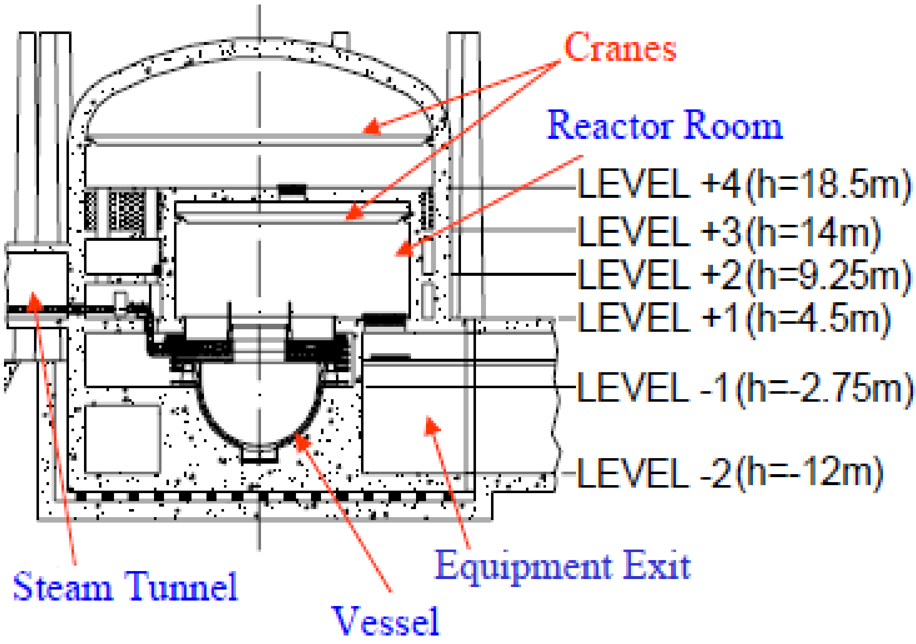
- Energy dissipation (due to relatively high viscous damping) to reduce lateral displacements;

- Sufficient stiffness at small displacements to provide adequate rigidity for service level environmental loads, e.g. wind and traffic-induced tremors.

In a base isolated structure the seismic protection is achieved by shifting its natural period away from the range of the frequencies for which the maximum amplification effect of ground motion is expected.
In this way the horizontal input seismic energy introduced into the structure is significantly reduced and consequently it is possible to avoid large plastic deformations and related damage phenomena due to non-linear response: high damping rubber bearing (HDRB) isolators have been considered in this study.




(a)




(b)

Figure 3 - Reactor Building main dimensions (a) and general configuration (b).

The use of a compact solution for the RV and a simplified and innovative primary circuit (key parameters of ALFRED reactor are summarized in Table 1), characterized by the possibility to remove all the internals, are useful to mitigate the possibly adverse effect of the high density of lead [7].

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 range is between the 480°C at the core outlet.

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

**Table 1 - Key parameters of ALFRED [6].**

| | |
|---|---|
| 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 (e.g. laminated or high damping rubber bearing) |

After that, as for the internals of reactor vessel concerned, the major portion of the fuel assembly is composed of wire wrapped fuel rods contained within a hexagonal duct (Figure 2). The fuel rods are of small diameter and are capable of sliding relative to each other and the duct. As a result, the contribution of the pin bundle to the assembly bending stiffness is negligible relative to the duct (Figure 4).
As indicated in Figure 4, the core load pad is simply a portion of duct having increased thickness.
The top load pad (TLP) is located on a transition section between the fuel rod bundle and outlet nozzle. This section is relatively thick and as a result the TLP is essentially rigid when lateral loads are applied. In contrast, the above core load pad is compliant and its characteristic stiffness depends on the nature of the applied/sustained loading (e.g. mechanical, dynamic loads, etc.).

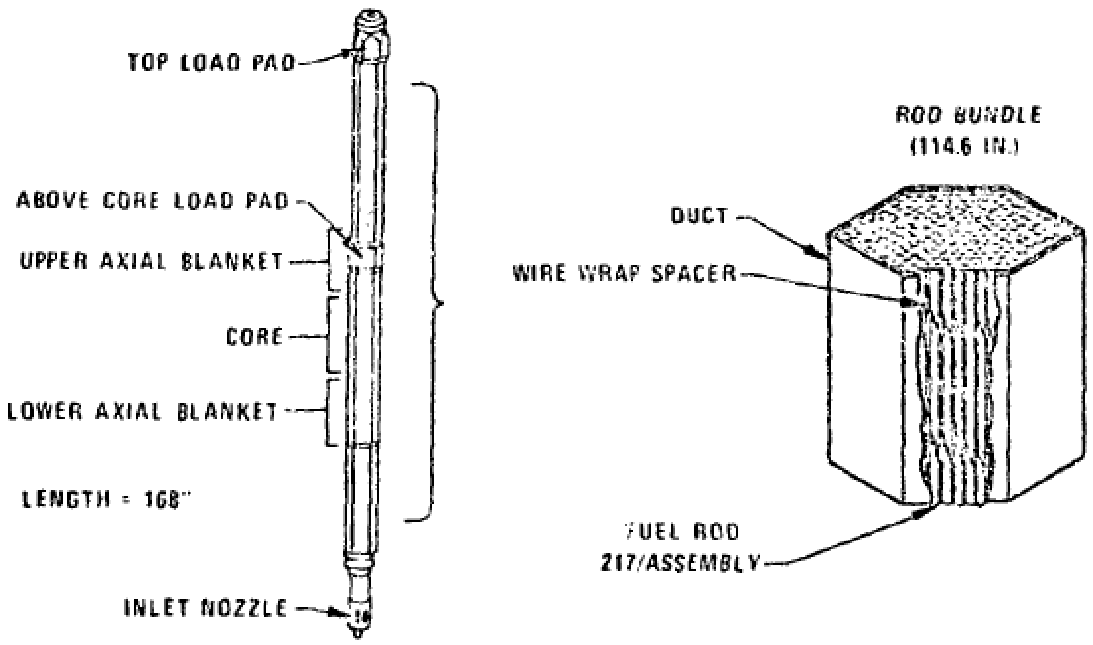



Figure 4 – General scheme of fuel restraints.

## 2.2. ALFRED core design

Specifically, the ALFRED reactor core is made of 171 Fuel Assemblies surrounded by a shroud of 108 Dummy Elements (Figure 5), in which 16 control elements are included [9].
Figure 5 shows a planar view of the arrangement of assemblies of the ALFRED core.

The active zone of the Fuel Assembly is yet defined and each hexagonal FA contains 127 pins; the main dimensions of which are reported in Figure 6.

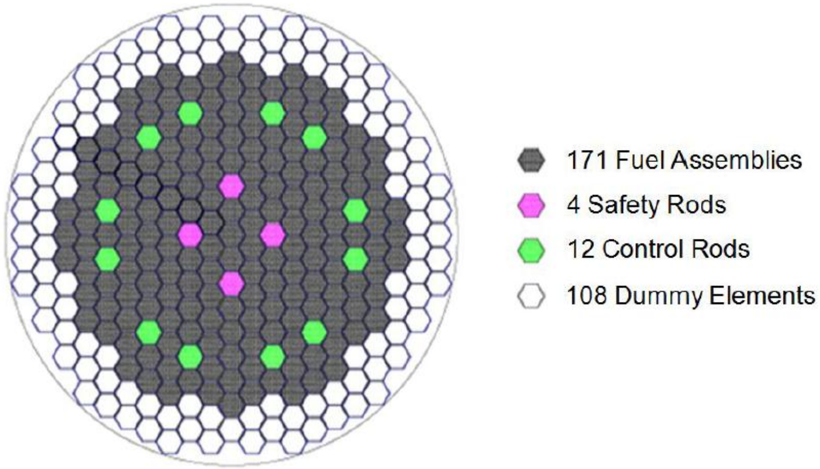
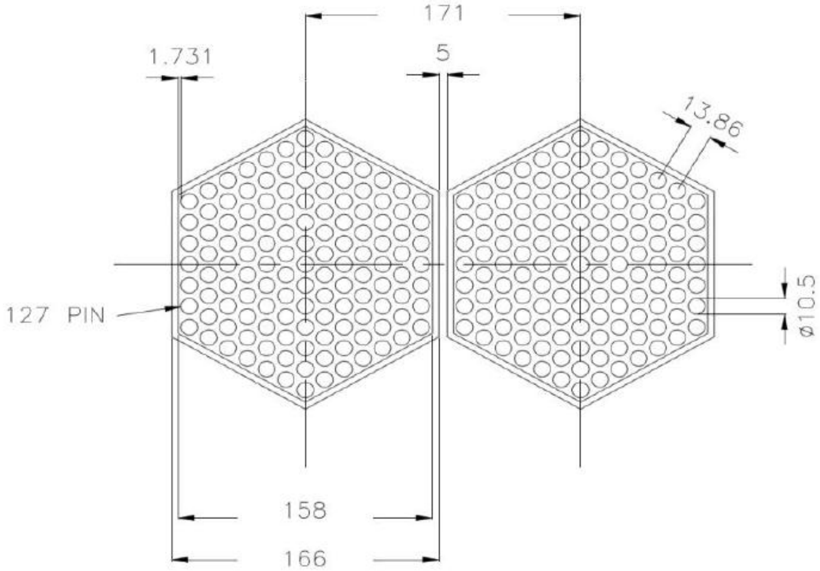



Figure 5 - ALFRED core configuration.

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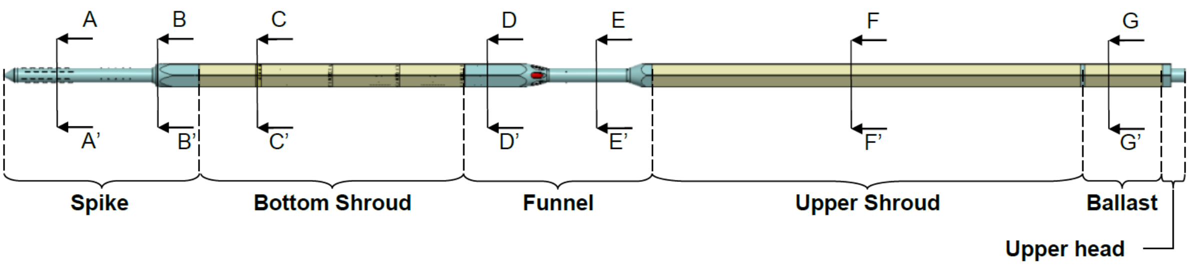



(b)

Figure 6 - Hexagonal FA geometrical reference data (a) and whole geometry (b) [9].

Moreover as indicated in [9], the FA structure of ALFRED is made up of the following parts (from the bottom to the top in Figure 6):

- The Spike, to guarantee the lead flow inlet both into the FA sub-channels and into the by-pass region between adjacent FAs.
- The Bottom Shroud, in which the core active zone is included.
- The Funnel, in correspondence of the outlet region.
- The Upper Shroud, which is the structural element above the outlet region, and it allows overcoming the lead free level.
- The Ballast, in the upper FA zone, required to maintain the FA in its position during the refuelling operation.
- The Upper Head, needed to connect the FA to the upper grid guaranteeing its correct position during normal operating condition.
  It has also to allow the connection of the FA to the refuelling machine during refuelling operations.

# 3. Core Compaction Modelling Strategies

Historically, two dimensional models of cores were used to design the LMFR assemblies against seismic loadings. These include models of a single row of assemblies as well as models of a horizontal plane of the core, as indicated in [10] [11]. Some of these models, such as the one described in Preumont (1987), did consider the fluid coupling created by the thin fluid layers between assemblies, while in the work of Martelli [12], the coupling effect of the fluid was replaced by an added mass on each assembly.
To model the fluid effects (by means of a 3D code), Moussallam et al. (2011) assumed the fluid effect as purely inertial, with no energy dissipation, and small relative displacement between adjacent assemblies: in this way the full Navier-Stokes equations describing the fluid behaviour are reduced to the wave propagation equation associated with a pressure boundary condition at the interfaces with structures [12].
More recently, several more complete models have been proposed by various authors ([13][14][15][16][17][18][19]), which consider also the fluid structure interaction effects: this approach will allow to determine in as much detail as possible the behaviour of NPP SSCs safety margins for credible initiator events and identify/optimize the core design features to limit the effects of dynamic solicitations and, in the same time, enhance the core resistance to even more severe scenarios.
The types of dynamic loadings to consider in the design of a core, are, generally, the following:

- Fluid-induced vibration excitation;
- Seismic excitation;
- Shock induced excitation;
- Internal energy release.

## 3.1. Fluid-induced vibration excitation

As for as the fluid vibration excitation concerned, the design must ensure that no damage could reasonably be induced by any fluid induced excitation (due to the coolant flow through the assemblies. Although the fluid-induced vibration excitation is cited here for reference, it is important to note that this phenomenon will not be treated in the present study, since its treatment requires a different modelling strategy than the one preliminary adopted for the other three types of loading.

## 3.2. Seismic excitation

The prevention of seismically induced damages has been a major concern in the design throughout the world. In doing that, some specific design criteria must be fulfilled that, from the assembly point of view, means: maximum allowable displacement, at the foot and head, maximum allowable acceleration at pins level, maximum allowable force on pads and maximum allowable core outer volume decrease (e. g. caused by the sloshing phenomenon

analysed by means of neutron analysis). Each of these criteria is of course related to the local behaviour and performance of the component analysed (for foot, pads and pins) or tested.
The cantilever behaviour of the assemblies, that may be assumed as elevated restrained structures, implies unavoidably high displacements at the bottom level (typically of the order of centimetre) as well as "high" moments at the top, where the flexural inertia is the lowest.
It is extremely important that this displacement must be kept below the value at which the control rod insertion could be prevented.
Impacts between assemblies at either pads or heads level may deteriorate their configuration and generate series of impulsive loadings on the pins, that could potentially lead to a breach into the first containment barrier.
*"… if pads are deteriorated during the event, a seismic excitation can induce variations of the core volume and an increase of reactivity which shall be kept below a certain level, sufficiently far from prompt-criticality phenomena (uncontrolled exponential increase of the fission reaction)"* [20].

## 3.3. Shock induced excitation

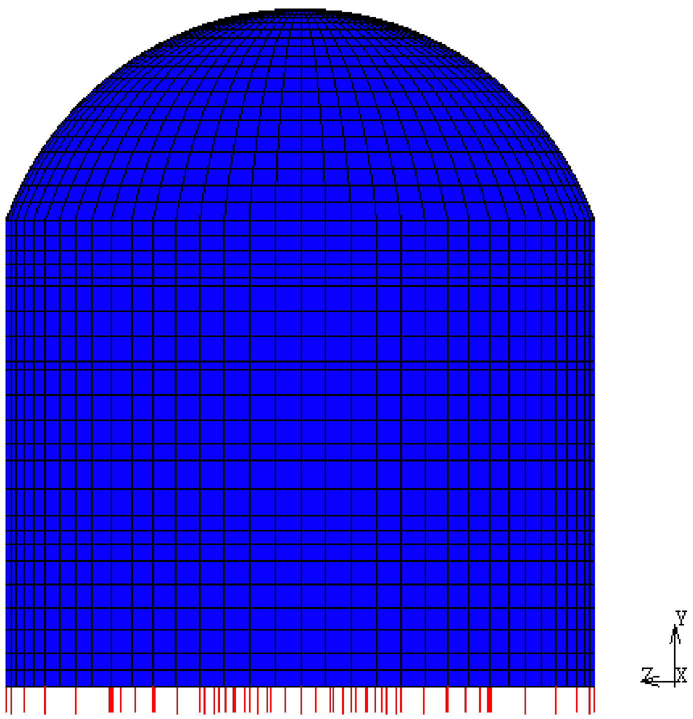
Such excitation can typically be associated generally to an accidental handling condition, to a possible steam generator tube rupture accidental event or to an external impact on the reactor building, typically the crashing of an airplane or the impact of a missile.
If the buildings are properly designed against such events, there are chances that they will be covered by the seismic design. Nevertheless in certain cases, as when the use of a seismic isolation system is foreseen, the seismic excitation can be brought down to a level below possible shock induced vibration. These loadings, of shortly duration, are characterized by an energy content concentrated in the high frequency range (far from the first assemblies' frequencies).

## 3.4. Internal energy release

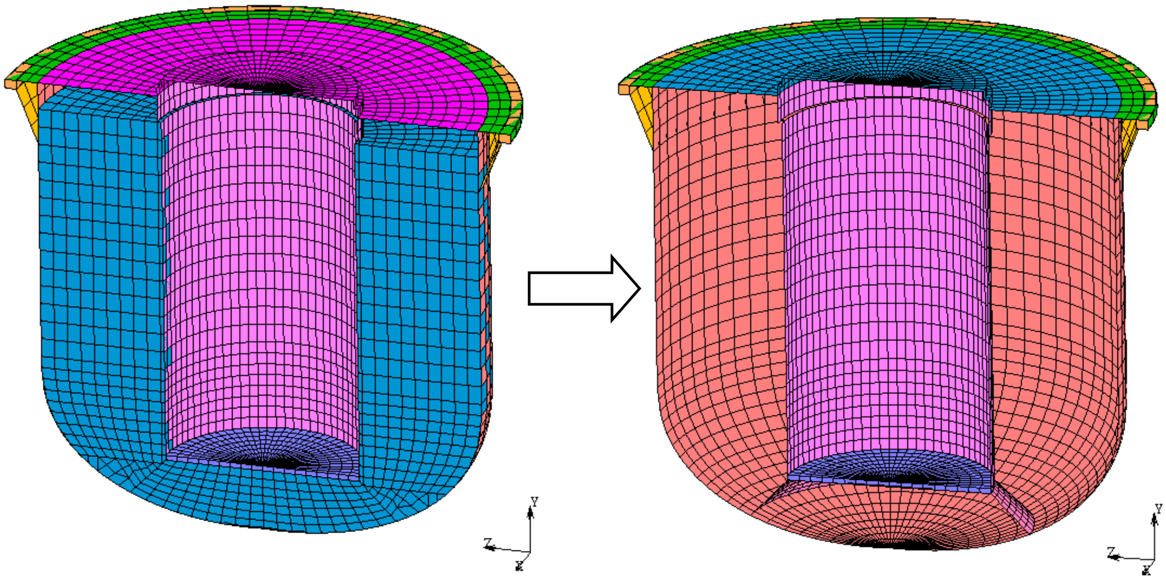
The internal energy release is not a design load case as seismic or shock induced vibrations. It represents a beyond design condition, potentially occurring in deteriorated situation that could also associated to/caused by a possible steam generator tube rupture accident, which should not induce a cliff edge effect in the safety demonstration of the reactor.
Namely, whatever the initiating event (either in normal or in accidental conditions), the nature of the energy release and its forms (pressure wave, kinetic energy of the coolant, etc.), it shall not bring the core to the limit of the prompt-criticality.

# 4. Approach to core compaction analysis

Core restraint analysis entails the modelling of the interacting behaviour of core assemblies and the bounding internals structures [4] [8].
The interaction among subassemblies, which occurs as the result of the contact between pad surfaces of adjacent subassemblies, makes the core deformation complicated. Therefore, it is important to consider the characteristics of pads in the analysis in order to attain detailed analysis.
Studies on this subject were carried out by Allbeson et al., 1984; Anderson and Maeda, 1990, etc..; they adopted a simple beam element to model a subassembly and did not consider detailed characteristics of pad portion, i.e. the additional bending moment due to the friction at pad surface, the change of compression rigidity of pad cross-section due to contact pattern at pad, the deformation of pad cross-section due to thermal expansion, internal pressure etc.
Reinhall et al. [21], instead, pointed the importance of the effects of the additional bending moment due to the friction and the change of bending moment rigidity due to the interaction between duct and fuel pins.
Because of its complexity and due to the lack of information related to the geometrical shape (design data and restraints characteristics), simulation of dynamic effects of the core with assembly interactions together was not carried out in the set up and implemented 3-dimensional models. Specifically the core region has been represented by means of its geometrical boundaries, through its top and bottom restraints, and by the fuel mass considered as lumped mass distributed on the fuel supporting plate.
The 3-D models, implemented by MSC©MARC code, showed in Figure 7, are used to obtain the definition of the assembly loads and displacements for further reference design calculations and verification of simplified core restraint models.




(a)




(b)

Figure 7 - Models of the Reactor Building (a) and RV (with and without lead) (b).

The structures and systems and components modelled are the RB, the Safety Vessel with its annular box structure, the Reactor Vessel and its support system; the inner cylindrical vessel and the fuel plate, that surround and sustain the core in the vertical position, and the pure lead. At this first stage of the analysis, the lead has been also considered as a structural material.
The inner cylindrical vessel wall was assumed 5 cm thick, while the fuel supporting plate was considered as a whole without holes.
The models were set up, assembled with appropriate elements, as for example the 3-D solid brick and/or shell thick elements, for dynamic/seismic analysis, available in the used finite element code. Moreover the viscous damping, that is generally less than 0.5%, was considered along with the structural one in the Rayleigh formulation.
With reference to the seismic excitation, the preliminary methodology used to determine the dynamic behaviour of the inner vessel/core region is characterized by the following steps:

1. Definition of the peak ground acceleration of the input earthquake, since effective isolation characteristics would be obtained by the chosen artificial seismic waves;

2. Definition of type and number of isolators based on a fixed frequency;

3. Determination of dynamic behaviour of RB isolated system, in terms of accelerations and displacement;

4. Determination and analysis of the global dynamic response of reactor vessel and its main components and systems as a result of the propagation of dynamic loads obtained at the point 3.

## 4.1. Criteria to select isolators

In the frequency domain of interest for most of the equipment, the response spectra are lower using base isolation, as widely addressed in literature. This means that seismic loads are reduced for standard components and the seismic safety margins, therefore, increase.
These margins are even more significant because the reactor vessel is in the upper part of building, where amplification of these loads may become significant (very great) for the conventional design approach.
Care should be taken, however, to design the piping between the nuclear island and a non-isolated building to accommodate the possible large horizontal displacements (by use, for example, of expansion joints or bellows) accompanied with by the RB rigid body behaviour.
The 3D isolators, HDRB type, have been represented by means of an iso-elastic approach made of spring and dashpot, indicated under the RB foundation, in the previous Figure 7, with red lines.
The number of isolators, appropriate to attain an isolation period of 2 s (f = 0.5 Hz) and in the same time to adequate support the vertical RB mass were calculated according to the procedure indicated in [18], while the stiffness of each isolator complies with the experimental data available in the FIP catalogue.

## 4.2. Seismic input

The seismic input used in the preliminary analyses of the core compaction is a synthetic earthquake acceleration time history which has a 24 s duration with 0.01 s time interval. The maximum peak acceleration of the time history has a zero period ground acceleration (PGA) of 0.3 g for sites with a low shear velocity, while it might be assumed equal to 0.2 g for sites with a high shear wave velocity.
The input acceleration data (Figure 8), applied at the base of the foundation of the isolated RB, were elaborated according to the updated Regulatory Guide US NRC 1.60 and 1.92, considering a 5% of critical damping value.

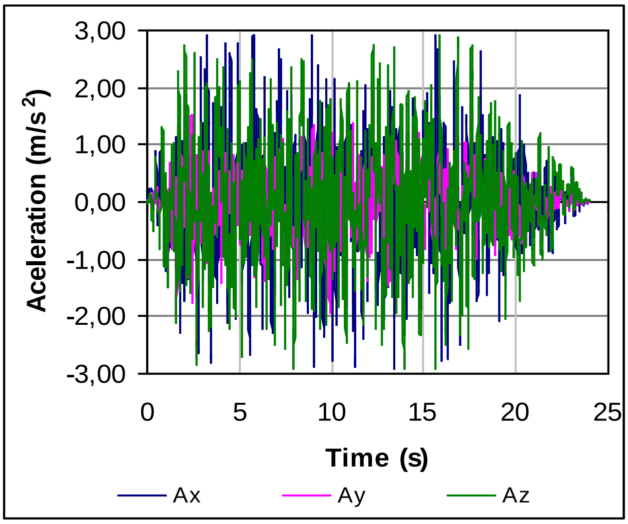
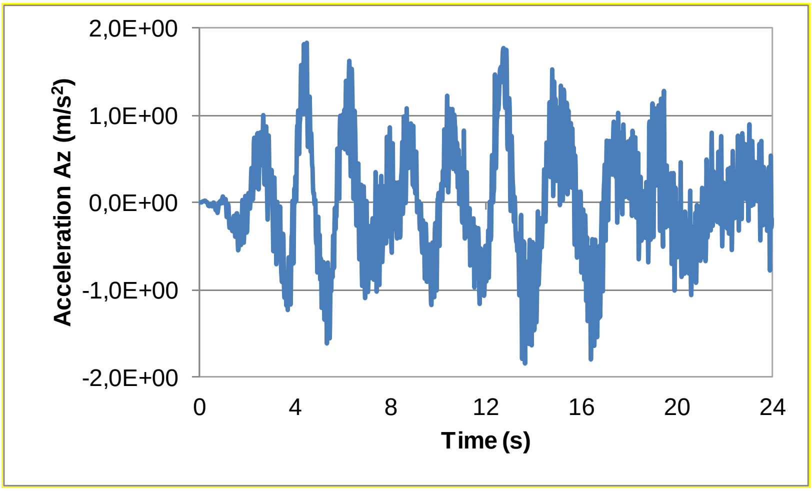



Figure 8- Seismic ATHs with PGA = 0.3 g.

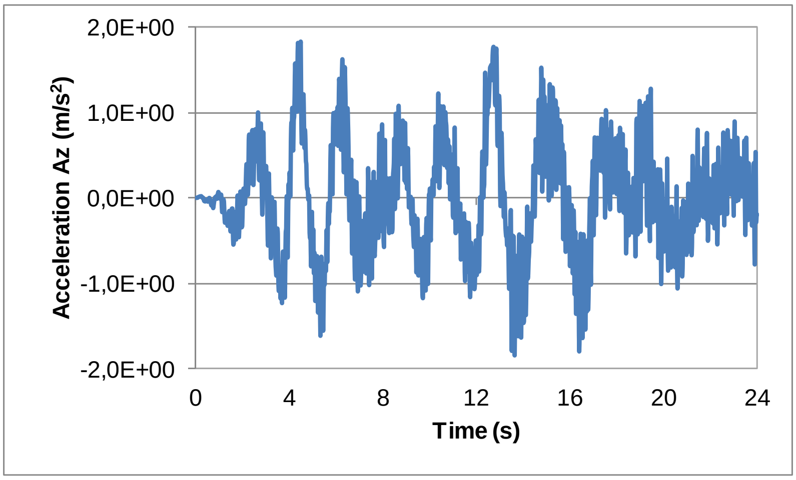
They were represented in previous Figure 8 by means of three artificial time histories components, two along the horizontal direction (Ax and Az) and one along the vertical one (Avert), compatible with the given free-field spectra, which represent the assumed DBE at a hypothetical embedment in stiff rock.
The input motion, in term of acceleration time history, was applied with all the three acceleration components in the three mutually orthogonal directions in order to obtain the propagation of seismic waves from ground up to the RV anchorage restraints.
Moreover the effects related to the possible amplification of the vertical acceleration component have been analysed.

# 5. Application to the ALFRED project

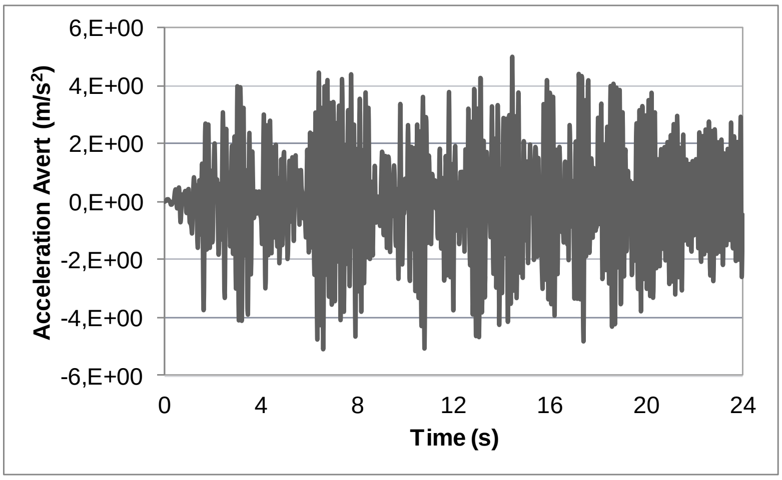
Before the evaluation of the effects due to the dynamic forces exerted/induced on to the RV inner structures, the influence of the dynamic loads propagating through the isolated reactor building was carried out. To the aim also a preliminarily modal analysis was performed to check the consistency between the isolated RB structure and the isolation system: the results obtained confirmed the RB structure behave as a “rigid body” (1$^{st}$ frequency of 0.5 Hz).
The acceleration values propagated up to the anchorage of the safety vessel, are represented in the diagrams of Figure 9: they confirmed the favourable effects of the isolation system in mitigating the propagation of the horizontal accelerations, of about 40%, inside and along the RB containment structure, while the vertical component is amplified along the building height.
The acceleration values, shown in Figure 9, were in turn used as input in the RV substructure (Figure 7 b) in order to analyze the structural effects induced by the ground motion on the RV internal components, particularly on the core region.
It is important to highlight that in this preliminary study the coupling effects between the fluid and the surrounding structures was not implemented.




(a)



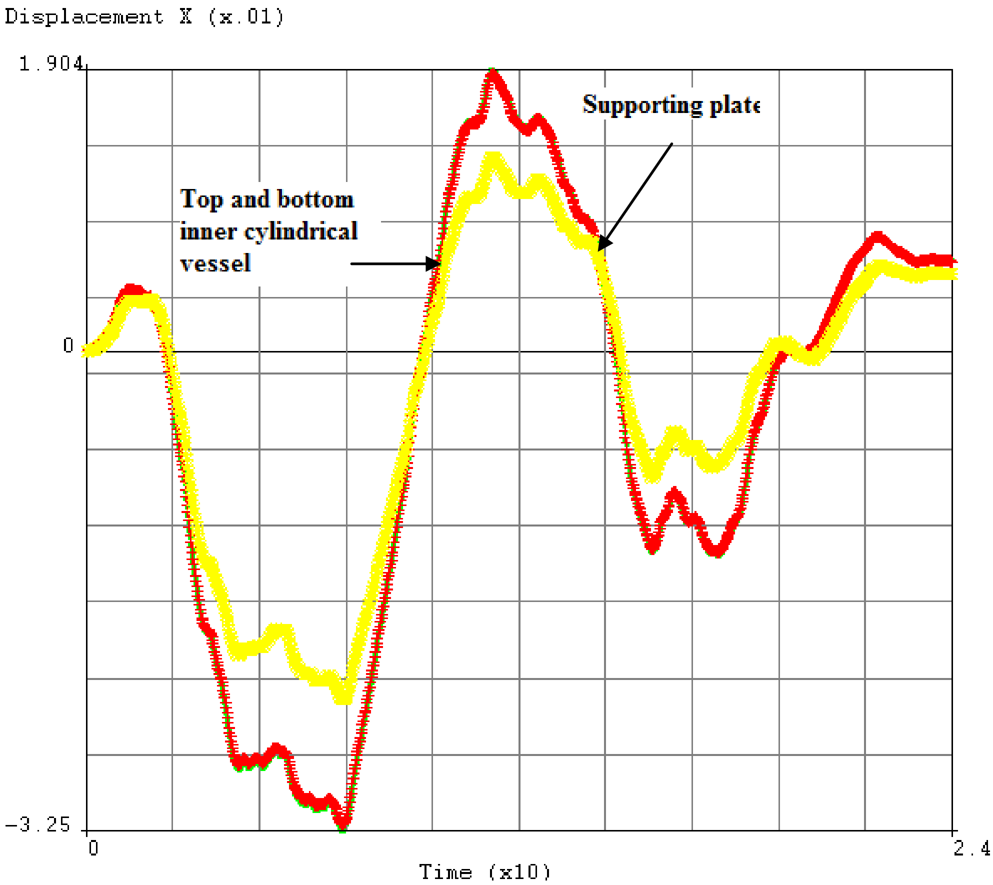

(b)



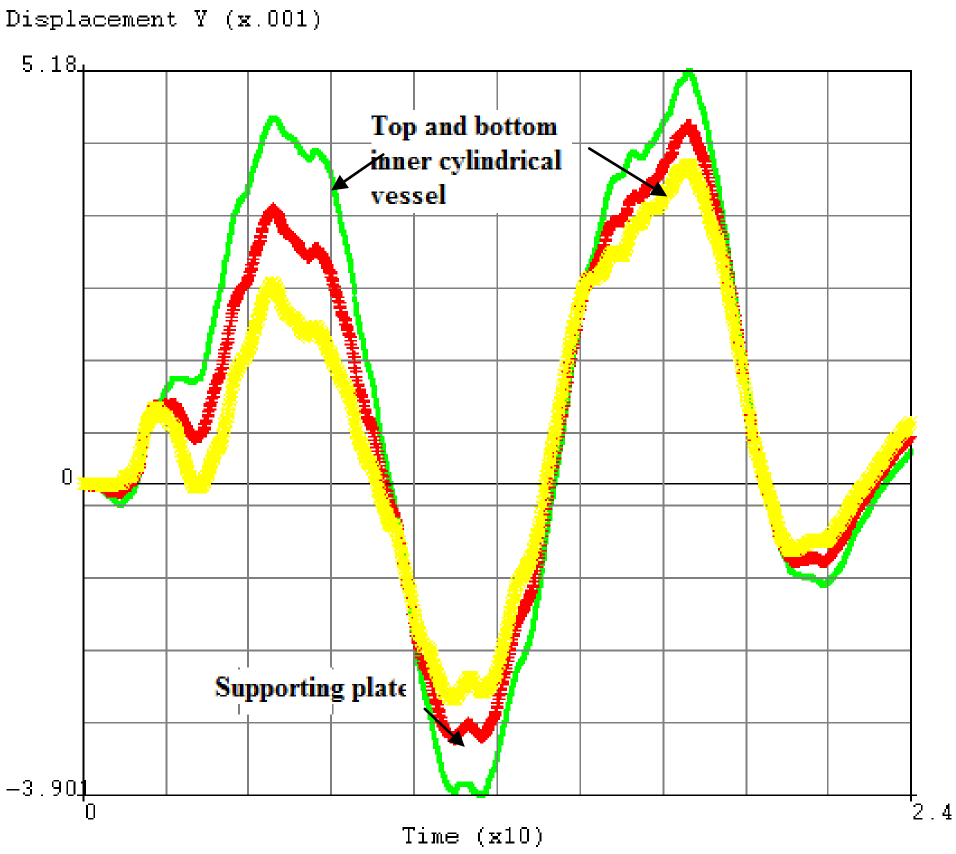

(c)

Figure 9 – Acceleration values propagated at the RV anchorage.

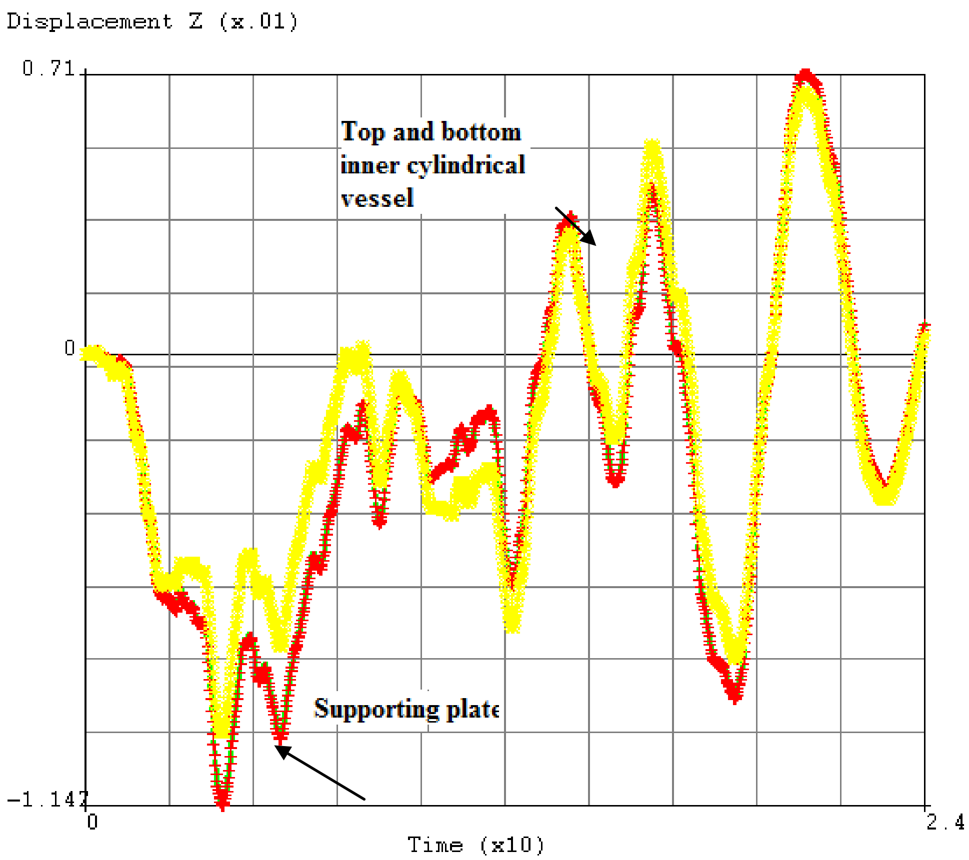
The results in terms of accelerations, displacement and stresses calculated at the top and bottom of the inner cylindrical vessel and at the fuel supporting plate were represented in the following Figure 10, Figure 11 and Figure 12.
These preliminary results highlighted a displacement in correspondence of the core region of about 2-3 cm.




a)




b)




c)

Figure 10 – Overall displacement at the core region.

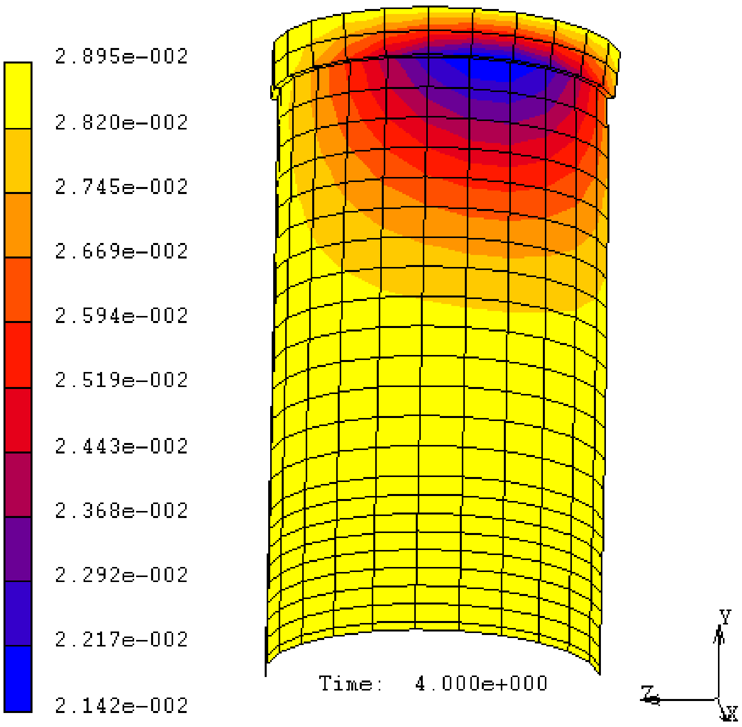
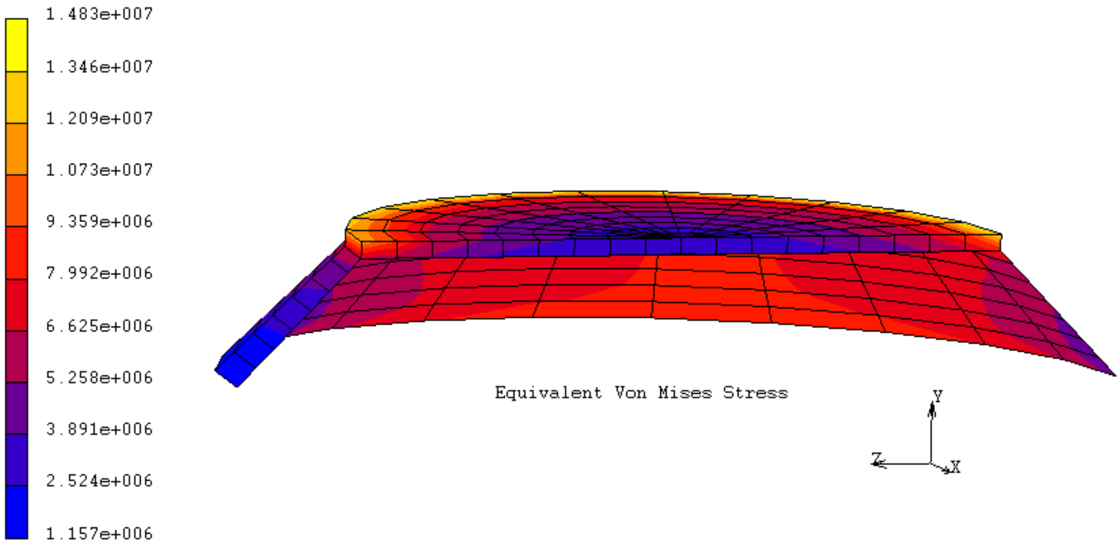
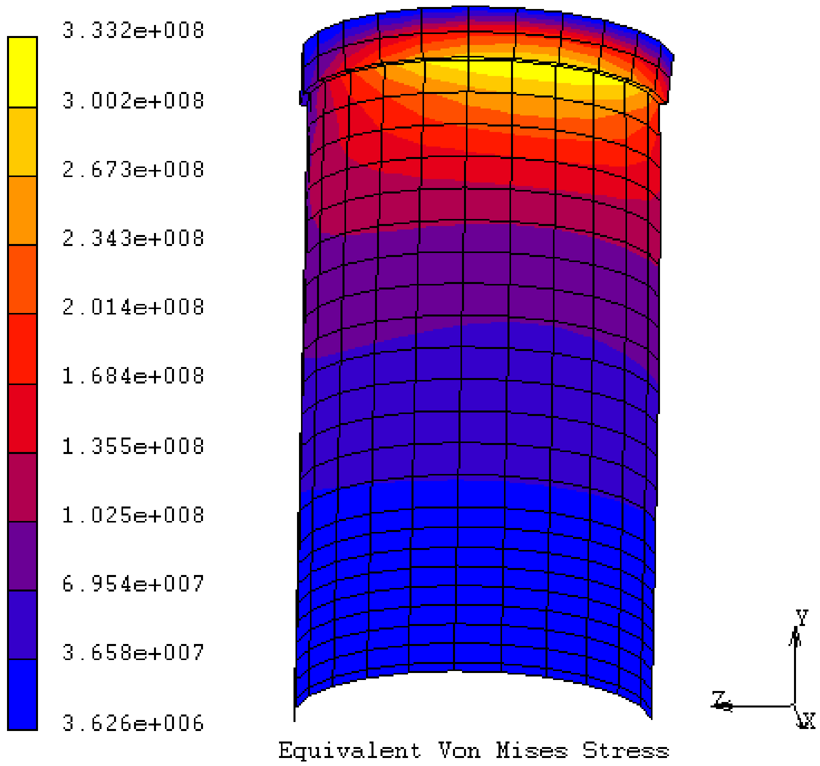



Figure 11 – Displacement overview at the core region at t = 4 s.




(a)




(b)

Figure 12 – Von Mises stress distribution in the fuel supporting plate (a) and inner cylindrical vessel (b) at t = 4.

Moreover the Von Mises stress overcome the yielding limit, particularly in the upper part of the inner cylindrical vessel. This means that the core region geometrical shape is undergoing deformation that might influence the normal reactor operation.
The stress distribution in the bottom part of the IC seemed instead to be mainly influenced by the boundary conditions of the support structure itself as well as by the assumption made of whole plate without holes.
Sensitivity analysis was also carried out to evaluate the influence of the mesh size and element type: the obtained results highlighted that these aspects did not affect a lot (about 5%) the dynamic response of the RV structure analysed.
Finally, these data will be used to analyse the fuel assembly and sub-assembly element in order to determine the deformation and level of compaction caused by such a type of dynamic load.

### 5.1. Future developments of the core analysis: characterization of the assembly and pads restraints

The key point of the core deformation analysis is how to catch the interaction among the subassemblies. Generally, subassemblies are modeled by means of beam elements in core mechanics.
As a total behavior a subassembly can be expressed by a beam structure, however, a wrapper tube of a subassembly behaves as a shell structure in the part of the whole structure.

Since contacts between subassemblies mainly occur at pads, it is natural that a much more attention should be focused on the characteristics of pads; in doing that a detailed geometrical characterization of assembly, subassemblies and of the core supporting plate, including also the restraints conditions are felt necessary to determine the phenomenon of the core compaction.
As illustrated in Figure 13, the deformation of the element/pad cross-section may be caused and influenced by the friction force, the contact force (responsible also of a change of the compression rigidity) and gap condition.
Due to the fact that these characteristics cannot be considered with beam and contact elements, therefore, the future developments of the present study is the implementation a suitable model of the fuel element with correct boundary conditions and characteristics/specification of the pads: a variation of cross-section flexibility may result in/influence the cross-section deformation.
Furthermore, to ensure the core structural integrity, an accurate dynamic analysis should be carried out take into account the non-linear behaviour especially related to the fluid-structure interaction.

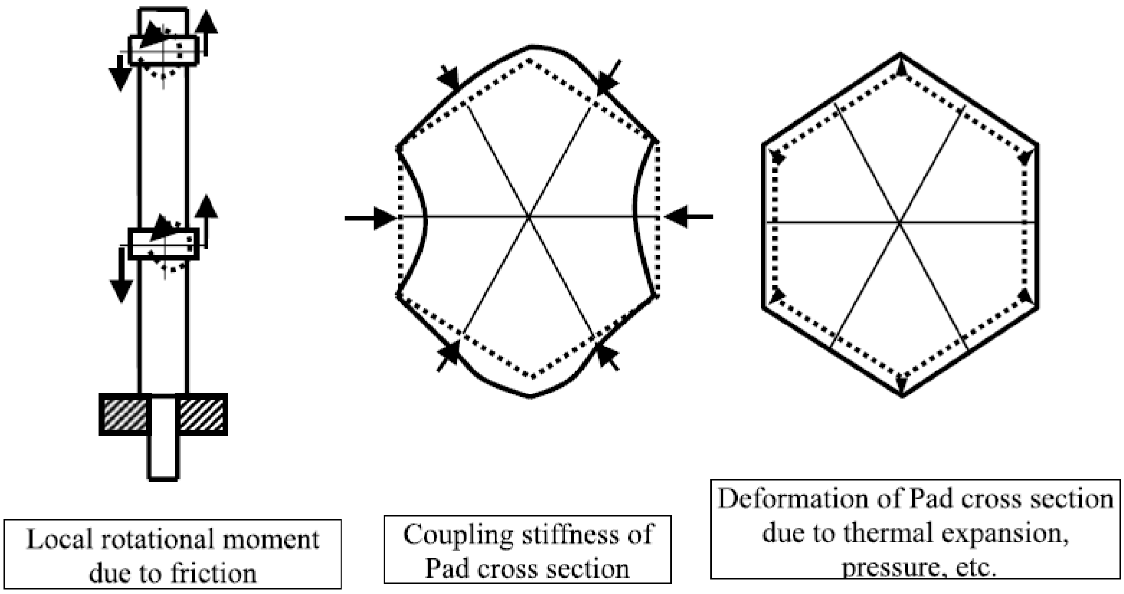



Figure 13 – Deformation phenomena of a subassembly [22].

# 6. Conclusion

In this report the results of a preliminary core mechanics analysis is discussed, as obtained using the Time History method coupled to the substructure approach, that allowed to study separately a hypothetical containment building and the reactor vessel with the inner cylindrical vessel.

The key point of the core deformation analysis is how to catch the interaction among the subassemblies. Because of the complexity and mainly due to the lack of information related to the geometrical shape, design data and restraints characteristics, the simulation of dynamic effects on the core was performed on a simplified 3-dimensional model, in which the fuel mass has been considered as lumped mass distributed on the fuel support plate.

The obtained preliminary numerical results, for the implemented models, highlighted that the overall displacement in correspondence of the core region resulted about 3 cm.

The Von Mises stress resulted to overcome the yielding limit, particularly in the upper part of the reactor vessel and inner cylindrical vessel. This means that the core region geometrical shape is undergoing deformations that might influence the normal reactor operation and result in a reactivity excursion.

Moreover the stress distribution in the bottom part of the IC structure seemed to be influenced by the assumed boundary conditions and structural flexibility structure itself.

Sensitivity analysis was also carried out to evaluate the influence of the mesh size and element type: the obtained results highlighted that these aspects did not affect a lot (about 5%) the dynamic response of the RV structure analysed.

Finally, these data will be used to analyse the fuel assembly and sub-assembly element in order to determine the deformation and level of compaction caused by such a type of dynamic load.

In conclusion it is important to note that future further developments should be necessary to evaluate more in depth the deformation of assemblies and subassemblies as well as of the element/pad cross-section versus the friction force, the contact force and gap condition in order to determine the level of compaction caused by the dynamic loading considered in input. Furthermore, to ensure the core structural integrity, an accurate dynamic analysis should be carried out take into account the non-linear behaviour especially related to the fluid-structure interaction.

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