



MASTER THESIS Energy and Nuclear Engineering

Analysis of reactivity effects due to core deformation in a Sodium-cooled Fast Reactor

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Academic Year 2018/2019

A chi mi ha sempre e sinceramente supportato.

Contents

1	Intr	oduction to the work	6
	1.1	Background	7
	1.2	Aim of the work	8
	1.3	Partnership	8
2	Fun	damentals of Sodium-cooled Fast Reactors	10
	2.1	Overview on fast reactors	10
		2.1.1 Fundamentals of physics of fast neutron reactors and re-	
		lated design features	15
		2.1.2 Advantages and drawbacks of fast reactors	21
	2.2	Sodium-cooled Fast Reactor Design	23
		2.2.1 Past, current and under development SFRs systems	26
		2.2.2 Past experiments on macroscopic structural changes of	
		SFR core	29
3	Des	cription of simulated core geometry and implementation of	
	defe	prmed geometries in TRIPOLI-4 [®] and Cast3m	32
	3.1	Description and composition of the reactor core model	33
	3.2	Implementation of deformed geometry in TRIPOLI-4 [®]	35
		3.2.1 TRIPOLI- $4^{\mbox{\tiny B}}$ code and settings of the simulations	35
		3.2.2 Definition of deformed core in TRIPOLI- $4^{\mbox{\scriptsize \$}}$	36
	3.3	Implementation of deformed geometry in Cast3m	42
		3.3.1 Mobile mesh method on Cast3m	42
		3.3.2 Definition of deformed core in Cast3m	42
	3.4	Specifications of selected deformed arrangements of the core	46
		3.4.1 Straight compactions and widenings of the core	46
		3.4.2 Deformation of the core due to a 6MJ explosion	48
		3.4.3 Flowerings of the core starting both from the central as-	
		sembly and from a more external one	49
4	Dis	cussion of the results	51
	4.1	Computational costs and convergence criteria of calculations of	
		$\operatorname{TRIPOLI-4^{\$}}$	51
	4.2	Computational costs and convergence criteria of calculations of	
		the solver of Cast3m	52
	4.3	Presentation of the results	53
		4.3.1 Compactions and widenings	54
		- 0	

	4.3.2 Deformations during the transient following a 6MJ exp	olo-
	4.3.3 Flowerings starting from the center of the core and fr a more external assembly	om
5 Con	clusion and suggestion for future work	
List of	Figures	
List of	Tables	
A Tab	es of reactivity and k_{eff} values computed with TRIPO	LI-4®
and	CAST3M	
A.1	Compactions	
	A.1.1 TRIPOLI-4 [®] results	
	A.1.2 Results of CAST3M using 15 and 23-point axial mesh	
A.2	Widenings	
	A.2.1 TRIPOLI-4 [®] results	
	A.2.2 Results of CAST3M using 15 and 23-point axial mesh	
A.3	Deformations recorded during transient following 6MJ explos	ion
	on top of the core	
	A.3.1 TRIPOLI-4 [®] results	
	A.3.2 Results of CAST3M using 15 and 23-point axial mesh	
A.4	Flowerings with source at the center of the core	
	A.4.1 TRIPOLI-4 [®] results	
	A.4.2 Results of CAST3M using 15 and 23-point axial mesh	
A.5	Flowerings with source located at the position of a fifth ring	as-
	sembly	
	A.5.1 TRIPOLI-4 [®] results $\ldots \ldots \ldots \ldots \ldots \ldots \ldots \ldots$	
	A.5.2 Results of CAST3M using 15 and 23-point axial mesh	

Chapter 1

Introduction to the work

In the actual global energetic scenario, where an increasing need of energy has to be satisfied according to the principles of increasing efficiency in the resources exploitation and of reducing carbon emissions and environmental impact, the nuclear energy has a prominent position among all supply options.^{[1][2]}

As a matter of fact, the nuclear power production technology does not produce carbon emissions and also nuclear energy has a very small life cycle carbon footprint resulting from the use of fossil fuels during mining, manufacturing and transport of materials and components.

Nevertheless, the capability of smoothing the integral of power required by the electric grid in favor of renewable (but intermittent) energy is the feature that allows nuclear energy to maintain a fundamental role in a climate-friendly future perspective. For this reason, nuclear capacity is considered to consistently raise by 2050, in order to avoid to overcome the limit global temperature increase of $2 \ ^{\circ}C.^{[3]}$



Figure 1.1: The 4 generations of reactor designs^[4]

In Figure 1.1 we can see the representation of the timeline of nuclear reactors development history: going from the 50's first prototypes of Generation I, passing through the worldwide diffused commercial nuclear power reactors technologies of Generation II (PWR, BWR, CANDU, RBMK, WWER) and arriving to advanced and evolutionary designs of Generation III (APWR, ABWR, WWER 1200, AP 600/1000, GT-MHR, PBMR).

Generation IV International Forum (GIF) is working on the development of a new generation of nuclear power plants (Generation IV) with the objectives of:

- increasing safety and reliability, reducing likelihood and degree of reactor core damage and trying to eliminate the need of offsite emergency intervention;
- reaching economic competitiveness through life-cycle cost advantages over other technologies;
- providing sustainable generation of energy, achieving effective fuel utilization and closed fuel cycle, minimizing the amount and facilitating the management of nuclear waste;
- enhancing physical protection (against acts of terrorism and natural hazard) and proliferation resistance.

Fast reactors represent a class of advanced nuclear reactors largely considered among Generation IV reactors designs, especially because of high efficiency in the use of uranium and capability of burning long-lived actinides. These features are fundamental to accomplish a closed fuel cycle and for these reasons particular attention is focusing on fast reactors design.^{[5][6]}

1.1 Background

The aim of GIF^[7] is to test the feasibility and performances of Generation IV reactors in order to prove that they could be commercially practicable. GIF R&D programs are focusing on these six different designs, identified in the GIF Technology Roadmap of 2002:

- Sodium-cooled Fast Reactor (SFR);
- Lead-cooled Fast Reactor (LFR);
- Gas-cooled Fast Reactor (GFR);
- Molten Salt Reactor (MSR);
- Very High Temperature Reactor (VHTR);
- SuperCritical Water Reactor (SCWR).

The roadmap, which is redefined about once per year by GIF, establishes indicatively the systems development timelines based on three phases: viability, performance and demonstration/deployment.

As reported in [8], GIF has 14 worldwide members (Argentina, Brazil, Canada, France, Japan, Korea, South Africa, Australia, Russia, Switzerland, Euratom, China, the United Kingdom and the United States) and each of them, except for Argentina, Australia, Brazil and the United Kingdom which are non-active members, participates in the development of one or more Generation IV systems.

France, through its implementing agency Commissariat à l'énergie atomique et aux énergies alternatives (CEA), is providing system arrangements concerning GFR, SFR and VHTR.

Phénix and Superphénix were two SFR prototypes of different sizes operating in the last decades of last century which ensured to France 45 reactor-years of operational experience and so a leading place in the know-how of this technology among the world. In 2018, CEA resulted involved in the development of two Generation IV nuclear reactor designs commissioned by french government: ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) and Allegro, an innovative gas-cooled reactor.

1.2 Aim of the work

The aim of this new generation of reactors is to increase their intrinsic safety and being capable to perform effectively sensitive analysis on these systems allows to perfect their design.

The objective of the present work is to analyze the neutronic effects of some structural deformations of the core, that could take place in a Sodium-cooled Fast Reactor.

Causes of deformations of a nuclear reactor core can be thermal expansion, irradiation consequences and also particular accidental transients.

Since deformations in the arrangement of the core impact on neutron transport, it is fundamental to know what is the reactivity response of the core in order to give explanation to eventual changes of power signal during the reactor operations. This is especially true for fast spectrum neutron reactors because of the important involvement of leakages in neutron balance, the low fraction of delayed neutrons and also because of the presence of large temperature gradients inside the core, which could cause strong thermal stresses in case of accidents.^[9] Uncontrolled compaction of the core is a safety and availability concern since it leads to a positive reactivity injection, while radial expansion gives rise to negative reactivity effects. These effects could produce unintended oscillating power transients.^[10]

In this thesis, we will examine different deformed configurations of an axially heterogeneous SFR core in order to :

- evaluate the corresponding reactivity changes;
- try to obtain a relation between the levels of compaction/expansion and their effects on the reactivity value of the system;
- demonstrate our capability in predicting the evolution of neutron balance in a core undergoing certain types of deformation.

1.3 Partnership

The results presented in this master thesis were achieved during a six-month internship from 14th May 2018 to 14th November 2018 in CEA-Saclay.

In particular, this internship was intended to be a collaboration between two departments: the Service d'Études de Réacteurs et de Mathématiques Appliquées (SERMA) in CEA-Saclay and Service de Physique des Réacteurs et du Cycle (SPRC) in CEA-Cadarache. SPRC provides cross sections, compositions and geometry features of the simulated core but also the deformed configurations which we will take into consideration in this analysis. In SERMA, I was able to exploit the experience of people who works on neutronic-mechanical coupling, on MonteCarlo method applied to neutronics, on the physics of fast neutron

reactors.

Since this work is the result of a partnership with a research centre, I am not allowed to spread some confidential information about the characteristics and design of the reactor on which I performed my analysis: for what concern the dimensions and compositions of the core assemblies, I will avoid to specify them or I will use fictive realistic values.

I take advantage of this paragraph to thank Guillaume Campioni, Cyril Patricot, Laurent Buiron, Jean-François Lebrat, François-Xavier Hugot and also professor Sandra Dulla for the support and for their technical supervising.

Chapter 2

Fundamentals of Sodium-cooled Fast Reactors

The object of this work is a pool-type Sodium-cooled Fast Reactor (SFR) core with the characteristic of having heterogeneous axial composition of the assemblies.

This chapter provides basic knowledge about fast reactors and Sodium-cooled Fast reactors in order to facilitate the understanding of the continuation of the work.

We firstly discuss the main features of fast spectrum neutron reactors, their advantages and their issues and then we will focus on SFR designs, attributes of some existing SFR prototypes, R&D goals outlined by the 2014 Update of the Gen IV International Forum (GIF) Technology Roadmap^[4] for this reactor type and mentions on some experimental studies on the neutronic effects due to modifications of the structure of a SFR core.

2.1 Overview on fast reactors

Fast reactors owe their name to the fact that fast neutrons (with energy higher than 1 keV) sustain the fission chain reaction. Consequently, there is no need of a moderator.

Fast reactors use relatively high-enriched fuel (around 20%) with respect to thermal reactors and its fuel also contains other fissile isotopes such as Pu-239 and Pu-241.

One favorable characteristic of FRs is the breeding capability; it means that they, while operating, can produce more plutonium than the amount of fuel (uranium and plutonium) they consume, resulting in having a better neutron economy with respect to traditional Light-Water Reactors (LWRs). As a consequence, the core power density in FRs is typically about five times the one in LWRs.

In this type of systems there is the chance to transmute long-lived actinides in shorter-lived ones, which allows to consistently reduce disposal time of spent nuclear fuel.

Fast reactors technology was already well-known in the 1950s and Enrico Fermi himself in 1945 said that "the country which first develops a breeder reactor will have a great competitive advantage in atomic energy".^[11] Uranium was thought to be not very plentiful resource at that time, so the development of breeder reactors were considered essential to pursue.

EBR-I (Experimental Breeder Reactor) was designed with the purpose to demonstrate the feasibility of a breeder reactor but it even went further becoming the first reactor to produce electric power in the world, in 1953. More than 20 of these reactors have been operating all over the world and some of them also supplying electricity, ensuring 400 reactor-years of operating experience.

FRs need a less-moderating coolant with respect to water used in LWRs.

In fact water consists in light atoms and according to conservation of energy and momentum, a particle striking with particles with pretty same dimensions loses faster its energy.

Therefore we can say that water coolant is very efficient in slowing down neutrons with respect to higher atomic number atoms such as metals like sodium or lead. For this reason the choice of the coolant is strictly dependent on the neutron spectrum we want to maintain in the reactor.

Fast reactor designs differ depending on the coolant employed:

- water above the thermodynamic critical point (374 °C, 22.1 MPa), which has such a low density to allow an hard spectrum;
- molten salt fluorides;
- gases composed by elements having low neutron capture cross section, such as helium or carbon dioxide;
- liquid metals or liquid metal alloys, such as sodium, lead or sodiumpotassium alloy.

The first coolant refers to fast supercritical water-cooled reactor which is under development in Japan, also known as Super Fast Breeder Reactor. Details about the core design of this system can be found in [12].

The attractive features of using a chemically inert coolant as helium in Gascooled Fast Reactors are that it does not dissociate and it cannot be activated inside the core, it is transparent (simplifying inspections and coolant handling) and it has low-moderating power so the coolant void coefficient is positive but small.

Moreover, being single-phase, the gas does not present phase transition-induced criticalities, like water does in PWR or BWR.

The gas allows to reach high operating temperatures (800-850 °C at the core outlet) which means reaching high efficiency of the power generation system, without incurring in corrosion or coolant radiotoxicity.

GFRs present the same advantages of fast-spectrum ones for what concern longterm sustainability, guaranteeing good exploitation of uranium resources and the reprocessing of spent fuel through the fission of long-lived actinides, and besides it has very high thermal efficiency (similar to VHTR) enabling indirect cycle driven by three IHX, using helium on the primary circuit, a Brayton cycle on the secondary circuit and a steam cycle on the tertiary circuit.



Figure 2.1: GFR design^[11]

One big challenge for researchers about GFRs is represented by ensuring decay heat removal in accidental conditions.

As a matter of fact, the low thermal inertia of the gas can lead to a rapid heating up of the core in case of loss of forced cooling and also the gas density is not sufficient to achieve natural convection for the core cooling, requiring very high value of power to the blower at low pressure.

Moreover severe irradiation conditions cause important effects on the core components which need to be considered in the design of the system.

Since 2005, research focused on fast-spectrum Molten Salt Reactors in which molten salt fluorides are contemporarily fluid fuel and coolant.

MSFRs open the possibility to exploit Th232-U233 cycle and to contribute to diminishing the radiotoxicity of spent fuel by lowering the amount of transuranic elements.

Fast MSRs present a unique characteristic: strongly negative reactivity coefficient, very important on the safety point of view. With respect to solid-fueled reactors, these systems contain smaller fissile inventories, they are less susceptible to radiation damages with the increasing fuel burnup, they do not need any special requirements in fabricating or handling solid fuel and they have an homogeneous isotopic composition of fuel.

As showed in Figure 2.2, MSFR power plant consists on three circuits involved in the power generation: the fuel circuit, an intermediate one and the power conversion one.



Figure 2.2: MSFR power plant scheme^[13]

Lead-cooled fast reactors (LFRs) operate using lead or a Pb-Bi alloy at atmospheric pressure and at high temperature; it is possible because of high boiling point of the coolant (about 1743 °C). These coolants have also the following interesting properties:

- chemically inertness;
- no exothermic reaction in contact with air or water;
- avoiding risk of core voiding due to coolant boiling;
- high density leading to fuel dispersion, not compaction, in case of core destruction;
- high heat of vaporization and high thermal inertia, important advantages in case of LOHS;
- shielding gamma-rays and retaining iodine and caesium, reducing the source of release of fission products in severe accidental conditions;
- low moderation property, which allows increasing of spacing between fuel rods reducing core pressure drop (making possible to exploit natural convection cooling in the primary system in case of shutdown heat removal) and risk of flow blockage.

Important drawbacks of this technology are erosion-corrosion effects due to high temperatures and high flow rates lead on structural steels, seismic/structural issues caused by the weight of the coolant, opacity of lead that makes challenging inspections and monitoring of reactor components.

A wide range of sizes of these reactors is under the attention of GIF: either small transportable system sized 10-100 MWe (Small Secure Transportable Autonomous Reactor or SSTAR - United States) reaching a 15-30 year core life or a system of intermediate size (BREST 300 - Russia) or a larger system rated at a commercial size, about 600 MWe (European Lead Fast Reactor or ELFR - Euratom). Representations of these designs are provided in Figure 2.3.



1 Core; 2 Steam generator; 3 Pump; 4 Refueling machine; 5 Reactor vault



Figure 2.3: LFR reference designs: BREST-OD-300 (top), ELFR (left) and SSTAR $(right)^{[11]}$

LFRs can also be employed for hydrogen and potable water production. LFRs design has typically a pool-type configuration and because of chemical inertness of the coolant there is no need of an intermediate heat exchanger system. Also this technology features a closed fuel cycle for efficient conversion of fertile uranium and management of actinides. Moreover, on the safety point of view, chemical inertness plus thermodynamic and neutron diffusion properties of the coolant enable to count on the possibility of using passive safety systems.

Sodium-cooled fast reactor, being the design on which we performed this analysis, will be the subject of Section 2.2.

2.1.1 Fundamentals of physics of fast neutron reactors and related design features

Fast reactors exploit the capability of fast neutron to produce fissions. For this reason they do not need a moderator and they use less-moderating primary coolants.

Fast reactor neutron spectrum (specifically SFR) is showed in Figure 2.4, compared to thermal reactor neutron spectrum (typical LWR).



Figure 2.4: Thermal versus fast reactor neutron spectrum^[14]

The FR neutron spectrum differs depending on the elastic scattering cross section of the coolant. GFRs, for example, present significantly harder neutron spectrum with respect to SFRs, because of lower neutron moderation or slowing down effects. The big difference between the two spectra is obviously also due to energy dependence of neutron cross sections of the fuel isotopic composition. This difference in the spectra produce a significant difference in the life-cycle of the fuel of these two kinds of reactors.

For this type of discussion, it would be useful to have in mind the energy dependence of capture and fission cross sections of principal fissile and fertile isotopes included in typical nuclear fuel mixture: U-235, U-238, Pu-239. Figures 2.5, 2.6, 2.7 represent fission and absorption cross sections of these 3 isotopes calculated at 300 K according to ENDF/B-VII.1 database in JANIS (Java-based Nuclear Data Information Software).



Figure 2.5: U-235 microscopic cross sections at 300 ${\rm K}$



Figure 2.6: U-238 microscopic cross sections at 300 ${\rm K}$



Figure 2.7: Pu-239 microscopic cross sections at 300 K

Natural uranium consists in 99.3% of U-238 and 0.7% of U-235. Comparing the values of fission cross sections of the two isotopes in the thermal energy range, we could easily understand that thermal reactors need enriched fuel to support the fission chain reaction.

It is important to notice that, while for low values of energy fission cross sections of the two uranium isotopes differs for more than seven order of magnitude, their difference is going to become lower and lower increasing the energy of the hitting neutron.

Fuel properties in a nuclear reactor are typically mixture of the ones of its composing isotopes, depending on the type of reactor (requested enrichment) and on the fuel burnup.

The capture-to-fission ratio (C/F), defined as the ratio between capture and fission cross sections at a certain energy of the incident neutron, allows to measure the "quality" of fissile isotopes. As a matter of fact, in order to sustain the chain reaction, the number of neutrons produced by fission should be higher than number of neutrons lost for capture or other mechanisms like leakage. Lower C/F ratio corresponds to more efficient fissile isotope.

As shown in Table 2.1, this parameter for Pu-239 at 100 keV is similar to the one characterizing U-235 at thermal energies, and at 200 keV it is even lower.

We choose to visualize the C/F ratio at these three energies because the first is typical of neutrons in thermal reactors, 100-200 keV is the mean energy of certain fast reactor designs, while 200 keV is about the peak energy that could be detected in a fast reactor. From Table 2.1 Pu-239 results in having lower capture-to-fission ratio at fast incident energy than U-235 at thermal incident energy.

Moreover, from Figure 2.7, we could see that for hitting energy of the neutrons higher than 100keV, the probability to produce fission of Pu-239 becomes larger

$E=0.025 \ eV$	$\sigma_c \ [barn]$	$\sigma_f \ [barn]$	C/F
U-235	99.5	590	0.169
Pu-239	272.5	752	0.362
$E=100 \ keV$			
U-235	0.448	1.550	0.289
Pu-239	0.248	1.552	0.160
$E=200 \ keV$			
U-235	0.310	1.355	0.229
Pu-239	0.188	1.515	0.124

Table 2.1: Capture-to-fission ratio of U-235 and Pu-239 at different energies according to JENDL 4.0

and larger than the probability to be captured by Pu-239.

This property makes Pu-239 the ideal candidate to be used as fissile isotope for fast reactors fuel. As a matter of fact, we can introduce the definition of η as number of neutrons produced per absorption, as below:

$$\eta = \frac{\sigma_f \nu}{\sigma_f + \sigma_c} \tag{2.1}$$

with

 σ_f : microscopic fission cross section;

 σ_c : microscopic capture cross section;

 ν : number of neutrons emitted per fission.

It is easy, looking at the figures above, to verify that Pu-239 η is higher than U-235 one.

The possibility of using ²³⁹Pu as major fissile isotope provides an advantage in terms of neutron balance for fast reactors with respect to thermal ones. This advantage is also due to the fact that the number of neutrons released by fission is higher for Pu-239 than for U-235 and it also increases for increasing energy of the hitting neutron, as we can see from Figure 2.8.

Also the fact that long-lived actinides (neptunium, americium, curium and other fissile isotopes of plutonium, forming through the irradiation which fuel undergoes in the reactor) have the same trends of fission and capture cross section of Pu-239, contributes to have a better neutron balance in fast reactors. This typical feature of fast reactors results in a further advantage because fission of these nuclides larger than uranium leads to their splitting in lower atomic number fission products, reducing the amount of radioactivity of the nuclear waste, rather than being accumulated as in thermal reactors.

It is fundamental to remind that despite U-238 fission cross section is very low with respect to U-235 both at high and low incident energy, it can capture a neutron and, passing through a beta decay, be converted in the fissile Pu-239. For this reason, Uranium-238 is referred to as fertile isotope.

Fast spectrum makes breeding possible inside FRs: the phenomenon of producing more fuel than consumed one.

If FRs are designed to produce more plutonium and other actinides than the uranium and plutonium they consume, they are called fast breeder reactors



Figure 2.8: Number of neutrons released per fission^[6]

(FBRs). In this case, conversion ratio defined as below

 $CR = \frac{\text{fissile material produced}}{\text{fissile material destroyed}}$

is greater than 1 and it is also called breeding ratio, BR.

The breeding ratio represents how much new fissile fuel a reactor is capable of producing during its operation.

Only sodium-cooled fast reactors can reach breeding ratio much greater than 1 (≈ 1.3).

Figure 2.9 represents U-238 capture cross section compared to Pu-239 fission cross section: we can see that for high energy spectrum the gap between the two quantities starts increasing.

Looking at the thermal energy range, we realize that in thermal reactors, Pu-239 undergoes fissions as soon as it is created due to the fact that Pu-239 fission rate is so much higher than the U-238 absorption rate (or Pu-239 production rate).

In typical fast reactor energy range, U-238 absorption rate still remains lower than Pu-239 fission rate but they are comparable.

For this reason FRs needs initial enrichment of the fuel higher than in thermal reactors, in order to reach criticality.

Usually fast reactor fuel contains from 20 up to 30% plutonium and the rest is natural uranium.

The high burnup in FRs, caused by the enrichment, results in high energy generation which requires very efficient cooling. Consequently this type of reactors has smaller fuel rods and it needs a coolant with good heat transfer properties. Since U-238 fission rate is not negligible at high energies, also neutrons coming from fission of U-238 should be counted in the neutron balance.

The bonus of neutrons occurring in FRs contributes in breeding new fuel for



Figure 2.9: Comparison between Pu-239 production cross section (blue line) and Pu-239 fission cross section (red line)^[6]

sustaining the chain reaction and in recycling depleted uranium of the "fertile blanket", which we can usually found around the core to maximize the breeding, transmuting it in pure Pu-239 or fertile Pu-240 (the blanket is then reprocessed to recover plutonium for being used in the same reactor or as fuel in other FRs). Long core life (which does not need refueling) is possible through breed-and-burn concepts.

High density of particles inside the core has as a drawback the stress due to high neutron irradiation which structural materials undergo in FRs.

Plutonium and Uranium are used as fast reactors fuel in the form of:

- oxides (UO₂-20PuO₂) with low thermal conductivity, low density of fissile materials, with helium-filled gap between the fuel and cladding but not reacting with liquid metals;
- metals (U-20Pu-10Zr) with very high thermal conductivity but undergoing swelling and melting at relatively low temperature compared to boiling temperature of liquid metals and not compatible with lead, due to solubility in case of cladding failure;
- nitrides (UN-20PuN) with high thermal conductivity, high density of fissile atoms but subject to swelling and C-14 contamination (from reaction N-14 + neutron);
- carbides (*UC*-10*PuC*) with high thermal conductivity and high density of fissile atoms but also highly subject to swelling and poorly compatible with air and water;
- fluoride-based liquid fuels of various compositions (solvent, fertile, and fissile) allowing operation as breeder or burner.

In particular oxide fuels have helium-filled gap between fuel and cladding, while metal alloy fuel pins have SS(316) or advanced alloy cladding, sodium-filled gap between fuel and cladding.

Metallic fuel ensures even harder spectrum because it is less effective in moderate neutron with respect to oxide fuel which contains oxygen, a fairly light atom.

On these two fuel designs, we already had irradiation experience in some FRs or test facilities.

Type of fuel is also linked to the fuel recycling technology which can be used. This process consists in processing a concentrate of metals (named "ore") to recover the desired mineral contained in it.

There are three kinds of metallurgical treatment which can be adopted :

- pyrometallurgy, using heat to trigger the separation between the metals and its concentrated mineral;
- electrometallurgy, using electric current to separate metals;
- hydrometallurgy, using acqueous solutions to dissolve the metal and sometimes also electrolytic cells to separate them.

Purex, an hydrometallurgical process, is the most currently used and it's the leading candidate for mixed oxide fuels.

Pyrometallurgical process is more suitable for mixed metal alloy fuels.^[15] Regarding control rods, in fast breeder reactors typical absorber materials used are boron carbide, europium boride and europium oxide. Their behaviour under high neutron irradiation (swelling, He-release) is the fundamental reason for their choice.

2.1.2 Advantages and drawbacks of fast reactors

In this paragraph we will resume principal characteristics of fast reactors we presented above dividing them in pros and cons of this technology. Advantages of fast reactors, especially if they are compared to thermal ones, are:

- 1. capability of breeding new fuel, solving the problem of "fuel shortage" because also depleted thorium and uranium can be bred and provide nuclear fuel, making more sustainable nuclear power;
- higher capability of transmuting transuranics (for their high fission cross section at fast energy neutron incident and the surplus of neutrons available in FRs core), splitting them in fission products that need significant less time to decay to harmless level of radioactivity, from hundreds of millennia to some centuries (see Figure 2.10);
- 3. breeding allows lower reactivity oscillations with burnup (need of lower excess of reactivity to control the reactor during its life and less reactivity available for accidental insertions);
- 4. possibility to exploit natural circulation of liquid metal coolant (high heattransfer properties) in LMRs and highly efficient heat transfer mechanisms of pressurized gas coolant in GFRs for the decay heat removal system, enhancing safety of the reactor;

- 5. benefiting from strong negative temperature coefficient (rising temperature results in decreasing in reactivity) which is an inherent safety property;
- 6. reduced parasitic capture and improved neutron balance allow flexibility of material selection (for example the use of stainless steels for structures).

Isotope	thermal spectrum	fast spectrum
Np-137	3	27
Pu-238	7	70
Pu-239	63	85
Pu-240	1	55
Pu-241	75	87
Pu-242	1	53
Am-241	1	21
Am-242m	75	94
Am-243	1	23
Cm-242	1	10
Cm-243	78	94
Cm-244	4	33

Figure 2.10: Transmutation probabilities expressed in percentages^[1]

Here below, we identified the drawbacks of these systems, on which research is focusing its efforts:

- 1. requirement of high enriched fuel which raises costs and risk of nuclear proliferation;
- 2. neutron activation due to high energy neutron flux which affects core components and the primary coolant;
- 3. fast time scales of operation (caused by smaller number of delayed neutrons in fast reactors) which allows power to vary faster than in thermal reactors for the same reactivity perturbation;
- 4. very long neutron mean-free paths lead to great sensitivity to neutron leakage and geometric core deformations;
- 5. problems related to coolants such as positive void coefficient (boiling of the coolant in accidental conditions would reduce coolant density and with it the absorption rate) and high reactivity with air and water to handle in SFRs and the large amount of power needed by the blower system to guarantee the cooling through forced circulation of the gas in GFRs;
- 6. high building and operational costs and as a consequence, low competitiveness with respect to thermal reactors unless a significant increasing in the uranium cost.

2.2 Sodium-cooled Fast Reactor Design

Sodium-cooled Fast Reactors or SFRs use liquid sodium at atmospheric pressure as coolant whose outlet temperature can reach 500-550 $^{\circ}$ C. This choice allows SFRs to operate at high-power-density and at the same time with low coolant volume fraction inside the core.

It is important to state what are the significantly favorable thermo-physical properties of Sodium:

- high boiling point;
- excellent heat transfer;
- high heat capacity;
- high thermal conductivity, being a metal;
- good compatibility with structural components and metallic fuels;
- thermal and radiation stability;
- negligible neutron moderation;
- low neutron absorption probability.

As consequences of these properties, on the safety point of view, the advantages of SFR technology are:

- low pressure primary and intermediate coolant system (which means low pumping power required);
- possibility of relying on natural circulation as passive safety system because of the presence of large variation of temperature in the core (so no LOCA concern and only guard vessel and guard pipes are needed to ensure coolant inventory);
- compatibility with conventional stainless steels;
- wide margin to coolant boiling point;
- negative feedback reactivity coefficient in case of increasing core temperature due to accidental conditions which guarantees inherent safety;
- low design containment pressure;
- simple operating conditions and accident management (long period of time available for corrective actions assured by huge thermal inertia inside the core in the case of pool-type SFRs).

Major disadvantages of use of sodium as coolant in FRs is that it chemically reacts with water and burns if in contact with air so it requires a sealed coolant system and also that it is a threat for the integrity of concrete. In some respects a liquid metal coolant is more benign overall than very high pressure water, which requires robust engineering on account of the pressure. However, the design needs to ensure that there is no chemical interaction sodium-water or sodium-air.



Figure 2.11: Typical LWR and SFR lattice schemes

The choice of the coolant leads to different arrangement of the lattice as we can see from figure 2.11.

In LWR water acts both as moderator and as coolant and the pitch to diameter ratio (P/D where P and D are defined in the figure) is a compromise to ensure together adequate moderation and sufficient cooling capacity.

In a SFR, so dium does not have to moderate the neutrons and thanks to its efficient heat transfer properties fuel pins is packed closer in a hexagonal lattice with triangular pitch. This compact arrangement of fuel pins means higher power density with respect to LWRs.^[16]

Fuel pins are typically separated through a thin wire wrapped around each of them.



Figure 2.12: Example of typical SFR fuel assembly structure (FFTF fuel assembly)^[17]

Loop-type configuration	Pool type configuration
More freedom in optimizing circuit components and in choosing their location	Radioactive primary coolant is all contained in the tank
Presence of simple boundary between hot and cold parts of the circuit	Higher thermal inertia and circuit arrangement reduce the impact of transients, components failure or leakages
Need of less structural support	No penetrations for control rods or refueling ports are needed
Containments for the reactor vessel	
(plus guard vessel) and for the	Combination of guard vessel and
primary cooling system rooms and	top dome as containment
double walled primary piping	
IHX can withstand higher pressure	
drop and its maintenance is easier	

Table 2.2: Advantageous features of the two SFRs configurations

Currently, two large size SFR configurations are taken into consideration: pool type and loop type.

Loop type reactors allows the sodium to leave the vessel. As a matter of fact the intermediate heat exchanger (IHX) is located in the containment building outside the vessel.

Pool type configuration consists in keeping the sodium confined in the reactor vessel, included the IHX.

For allowing thermal expansion of a liquid closed in a vessel there is a space between the level of sodium and the top of the tank filled with an inert gas (typically argon). The gas is maintained at slightly over atmospheric pressure to prevent any leaks from the outside of the system.

The coolant tank is surrounded by a leak jacket, to avoid that, in case of a leak, the sodium level would fall to the point that core would be exposed.

Moreover, sodium choice as coolant allows to realize small size (50 to 150 MWe) modular supported by a fuel cycle based on pyrometallurgical processing in facilities integrated with the reactor.

In Table 2.2 we report a comparison between all distinctive features of the two configurations.^{[17][16]}

For safety reasons, to avoid to have radioactive sodium in the boiler and to assure that no water or steam could react with primary coolant, SFRs have generally three heat transfer systems (as shown in Figure 2.13):

- primary coolant system, which cools the core;
- intermediate coolant system, which transfers heat from primary loop to the steam generator;
- energy conversion system, which generates electricity through the turbine.

Primary and secondary heat transfer systems are low pressurized (near atmospheric pressure).



Liquid Metal cooled Fast Breeder Reactors (LMFBR)

Figure 2.13: SFRs configurations: pool type and loop type^[16]

Sodium enters the core at about 400 $^{\circ}$ C and it is forced to pass between fuel pins. Hot sodium at about 550 $^{\circ}$ C flows down through tubes towards IHX where it transfers heat to secondary coolant. Secondary sodium then goes to the steam generator.

The power generation system is similar to PWR one but in SFR it runs at higher temperature reaching an higher thermal efficiency ($\approx 40\%$).

In SFRs, in addition to mechanical pumps, it is possible to use electromagnetic pumps (with no moving parts) because sodium is a metal with very high electrical conductivity.

Generally for IHX, shell-and-tube heat exchangers in counter flow are used.

Another system that we can find in the BOP is the Decay Heat Removal System and it is based on diverting steam from the turbine to heat sink through a bypass circuit; in the case this system is not available, it is possible to rely on passive heat removal mechanisms using natural circulation.

2.2.1 Past, current and under development SFRs systems

As discussed in the opening of Section 2.1, fast reactors technology has been pursued since the 50's. From that moment, a lot of experimental and prototype fast reactors was built and operated all around the world.

Table 2.3 shows all SFRs ever become critical.

Much of SFRs basic technology was already established in former fast reactor programs, but its feasibility was effectively confirmed by Phénix end-of-life tests in France, lifetime extension of BN-600 in Russia, the restart and success of core confirmation tests of Monju in Japan and the start-up of an experimental fast reactor in China. All of them represents nuclear historic landmarks.

Facility	1st Critical	Country
EBR-I	1951	USA
BR-5/BR-10	1958	Russia
DFR	1959	UK
Fermi	1963	USA
EBR-II	1963	USA
Rapsodie	1967	France
BOR-60	1968	Russia
SEFOR	1969	USA
KNK-II	1972	Germany
BN-350	1972	Kazakhstan
Phénix	1973	France
PFR	1974	UK
BN-600	1980	Russia
FFTF	1980	USA
JOYO	1982	Japan
FBTR	1985	India
Super-Phénix	1985	France
MONJU	1985	Japan
CEFR	2010	China
BN-800	2015	Russia
PFBR	2015	India

 Table 2.3: Sodium-Cooled Fast Reactors World Wide Experience

For our discussion, we will focus on european SFRs projects, especially french ones.

Rapsodie was the first experimental sodium-cooled reactor of France to become critical in 1967. Originally designed to be 20 MWth size system, its power reached 40 MWth in 1970 and then brought to about the same initial value until the shut down in 1983. It was a loop-type reactor with primary and secondary sodium loops. Its design was as close as possible to the basic one envisaged for the commercialization (molten sodium as coolant, reactor materials, the value of the power density, etc).

Phénix construction at Marcoule began concomitantly with the first year of operations of Rapsodie and it operated from 1973 to 2009 (specifically it operates the last months as research reactor for irradiation and transmutation of nuclear waste studies). It was a typical pool type SFR with primary sodium outlet temperature of 560 $^{\circ}$ C with three differentiated secondary loops (and steam generators) and with a turbine exploiting the more efficient Hirn cycle.

Phénix was supposed to produce 563 MWth but actually it underwent a long period of stop of the operations due to repeated shutdowns, caused by negative reactivity transients, and the consequent phases necessary for inspections and tests to guess the physical reasons of these episodes and for the refurbishment of the system (1990-2002). At the restart in 2002, the decision of reducing the thermal power output to 345 MWth was made.

Phénix operation conditions were oriented to maximize fuel burnup; all pluto-

nium produced in the reactor was recycled back in it after reprocessing. Phénix experience, including reprocessing high burn-up fuels, waste confinement, closed fuel cycle realization, demonstrates that fast breeder reactor could represent an industrial reality. The system was planned to reach breeding ratio of 1.13, but it actually overcame this value and it was able to produce 16% more fissile fuel than the consumed one.

Super-Phénix was built on the model of Phénix; it was a 1250 MWe prototype. It was the world's only commercial-sized breeder reactor. It operated only for 13 years since 1985 but, during its history, it experienced long shutdowns caused by significant incidents, consequent repairs. It was shutdown more than half the time of its operation and its total capacity factor, calculated as the number of kWh that it generated divided by the kWh that it was supposed to generate operating continually at full capacity, was less than 7%.

By the way, in the history of this type of systems, only Russian BN-600 recorded a decent capacity factor and this is due to the willingness of its officers to proceed with the operations despite multiple sodium fire accidents.^[11]

Regarding the future of SFR technology, R&D efforts of Generation IV International Forum are focusing on these challenges:

- increasing temperature operation and thermal efficiency (also through advanced cycles for energy conversion and innovative components design);
- optimizing fuel design, its interaction with the cladding and its compatibility with the sodium;
- sodium void worth which can be positive for large cores;
- shielding is complex to realize in fast neutron reactors;
- improving core inherent safety;
- avoidance of air-sodium and water-sodium dangerous reactions through efficient leak-tight system design and mitigation of their impact on safety system component through the use of inert cells, double tubes and steel liner;
- development of fuel handling technologies in order to facilitate in-service inspections and maintenance, which is very challenging due to opaqueness of sodium.

Resuming these concepts, a lot of R&D projects, usually consisting in collaborations between GIF countries, have as targets the improvement of economic performance and the insurance of robust behaviour in off-normal conditions (exploiting inherent safety mechanisms).

ASTRID (or Advanced Sodium Technological Reactor for Industrial Demonstration) was one of GIF major projects, commissioned to CEA supported by EC's European Sustainable Nuclear Industrial Initiative (ESNII) and Japan (since 2014). This system was expected to be a 600 MWe prototype of a commercial series of 1500 MWe SFRs using depleted uranium as fuel and burning the plutonium in used MOX fuel.

The design envisaged 25-35% plutonium enrichment, assurance of negative void

reactivity in the core and the possibility of using nitrogen with Brayton cycle gas turbine technology in the power conversion system.

To reduce the probability and consequences of severe accidents, the current design involves four independent heat exchanger loops.

ASTRID is going to fulfill all the criteria stated by Generation IV International Forum in terms of safety, economy and proliferation resistance.

CEA worked on fuel and core designs while AREVA (now Framatome), a french industry, supported it for the design of nuclear steam supply system, the nuclear auxiliaries and the instrumentation and control system.

ASTRID program also included associated fuel cycle facilities: a dedicated MOX fuel fabrication line and a pilot reprocessing plant for the used fuel.

In 2018 French government retracted his support to the 600 MWe ASTRID project. Then CEA proposed to rescale the size of ASTRID passing to 100 MWe, in order to reduce construction costs and building time. In September of 2019, CEA announced that it's no longer going to build ASTRID prototype in short or medium term.

2.2.2 Past experiments on macroscopic structural changes of SFR core

As shown in Figure 3.3, in fast spectrum reactors, thin fuel pins are closely packed in an hexagonal arrangement that represents the single assembly.

Typically there are fuel, control, reflector and shielding assemblies in the core of a fast reactor. All the assemblies are organized pursuing a bigger hexagonal pattern with a thin layer of sodium between them.

For the nature of the neutronic interactions inside a fast spectrum reactor core, FRs result very sensitive to structural deformation of arrangement of fuel assemblies.

Assemblies deformations could be the consequence of a combination of factors: high temperature, high neutron flux and coolant pressure difference.

The movements of assemblies can be triggered by seismic events or assembly bowing due to thermal or irradiation-related stresses.^[18]

Specifically thermal gradient could cause assembly bowing transients (elastic thermal strain) while fast neutron flux gradient can lead to assembly permanent bowing (inelastic irradiation creep and swelling strains).^[19] These phenomena become especially relevant in case of loss-of-flow (LOF) transients, because the power to flow ratio (P/F) reaches about the double of the value in nominal conditions.

In order to guarantee proper control of fuel rods and fuel handling operations, it is fundamental to maintain alignment of the assemblies.

If fuel assemblies are pushed closer together, the reactor experience a positive insertion of reactivity and viceversa, in case flowering of the core (assemblies moving apart) has the reverse effect. These reactivity feedbacks strongly depend on the way the core assemblies are supported and restrained.

Regarding core deformation, to provide inherent safety, Core Restraint System is deployed in the designs of Sodium-cooled Fast Reactors. Its objectives is to control radial movements of the assemblies caused by the reasons exposed before, to assure acceptable reactivity effects and alignment of control-rod driveline with adequate tolerance. Moreover it is necessary to provide sufficient clearances for easing refueling but also in order to accomodate eventual horizontal seismic motions within certain alignments and stress values.

Two different core restraint systems have been designed: free-flowering restraint system and limited-free-bowing restraint system. The first consists in two supports for the assemblies at lower grid plates, allowing their interactions at the Above Core Load Pads (ACLPs) and at the Top Load Pads (TLPs); the second includes also a restraint ring at the height of TLPs in order to control radial displacements of the outer-most assemblies. In Figure 2.14 we can see the representation of a limited-free-bowing restraint system. In these systems, some



Figure 2.14: Core restraint system details^[18]

parameters can be modified in order to reach the desired effect in different core designs:

- layout and stiffness of lower support adaptors;
- number, location and configuration of load pads;
- rigidity of peripheral belt.

To provide a proper core restraint system design, it is fundamental to be able to predict fuel assemblies motion over fuel life-cycle time and following reactivity effects, understanding the reasons underneath.

During the last decades, a lot of studies about core distortions, inter-component contact loadings, sodium-void effects, bowing and swelling phenomena in Liquid Metal Fast Reactors and their reactivity effects, tests on different designs of core restraint system were carried out all over the world.^{[18][19][20][21]}

Studies, similar to ours, involving the analysis of neutronic effects of geometric deformation of SFRs cores, was useful to achieve our results and helped us to understand them and they will be cited in the following chapters.^{[9][10][22][23][24]}.

Especially in [10], they performed some core flowering tests on Phénix through the use of "DAC" subassembly, a moderated experimental carrier. It is used to produce sodium boiling and the consequent vapor bubbles collapse induces core flowering and negative reactivity insertion. Phénix end-of-life test campaign was exploited in order to carry out this study on a irradiated core because of the availability of all experimental data needed for the calculations. The interesting results of this research are:

- when the source of the flowering is located at the center, the computed reactivity evolution is linearly related to the amplitude of the flowering (as long as the thickness of the sodium gap between the assemblies is not significant);
- the influence of initial reactivity value (supercritical or subcritical) is negligigle in this type of analysis;
- when the flowering source is located on the edge of the core the reactivity effect is still negative but the size of the reduction is lower than central source tests.

Chapter 3

Description of simulated core geometry and implementation of deformed geometries in TRIPOLI-4[®] and Cast3m

In order to calculate the reactivity changes due to mechanical deformations of the core, it is possible to use two different approaches: applying perturbation theory or direct criticality calculation on the deformed geometry.

The perturbation theory is used to calculate the reactivity effects due to small perturbations of the reactor core and its first order approximation allows to define the variation of reactivity between the perturbed and nominal configuration as function of adjoint flux of initial state, perturbed state flux and on perturbations of multiplication and absorption matrices, as described in [9].^[25] Using the other approach, the deformation can be represented by alteration of material density method, mesh deformation methods or mesh projection methods. These calculations can be made with either deterministic or Monte Carlo codes.^[9]

In this chapter we will firstly describe the core geometry on which this study has been performed.

The assumed SFR core configuration consists in an axially and radially heterogeneous design of a large size power reactor. Details about assemblies dimensions and axial compositions will be avoided or fictitiously attributed, as discussed in Section 1.3.

Thereafter we will briefly discuss about the two methods and codes employed for this analysis and then we will explain how the deformed geometry was implemented within the two codes.

3.1 Description and composition of the reactor core model

Tipically Sodium-cooled Fast Reactor core is a radially heterogeneous ensemble of axially heterogeneous hexagonal assemblies. Inside the core we will find different types of assemblies which have different functions: fuel assemblies, control-rods assemblies, diluent assemblies, reflector assemblies and shielding assemblies. In Figure 3.1, a typical SFR core like the one on which we performed our criticality calculations is represented.

Our core model consists in 15 rings of assemblies around the central one, separated by a thin layer of sodium coolant.



Figure 3.1: Example of typical SFR core design^[21]

The peculiarity of this core model is the presence of 3.10 meter long axially heterogeneous assemblies. This length does not contain the lower support with the coolant inner ports and the upper part with handling socket (see Figure 3.3 to have an example of the structure of an assembly).

The pitch, distance between the centers of two adjacent assemblies, measures 16.10 cm and it includes the layer of inter-assemblies sodium of 0.25 cm.

In Figure 3.2, we reported the axial description of all 9 different types of assemblies, which we had to reproduce in the two codes, in the order we encounter them going radially from the center of the geometry to the external boundary. Some of them (a type of control-rod, diluent, shielding and reflector assemblies) are axially homogeneous, while the others have an axially heterogeneous composition.

The assembly results axially heterogeneous if the pins, that compose it, present an axially heterogeneous structure.

It is important to notice that the fuel assemblies which stand in the inner part of the core have a different structure with respect to the ones in the outer part. In Figure 3.3, simplified schematic representation of inner fuel pins and outer fuel pins is presented.

Inner core fuel pins differ for having a central UO_2 fertile zone between two

 $(U, Pu)O_2$ fissile areas while outer ones has a 10 cm longer only fissile consisting active core length. These different configurations have been designed in order to achieve the objective of Generation IV reactors of realizing an intrinsically safe behaviour of the system in accidental conditions. Specifically, an heterogeneous structure is necessary to obtain null or negative sodium void worth, while these two fuel pin designs enhance minimizing secondary sodium activity level, preventing reactivity insertion in case of core compaction and reducing manifacturing costs.^[26] As we will see in the next sections, this heterogeneity of our model geometry will lead to serious complications in choosing the proper axial mesh for the definition of the core and for the description of the assemblies movements.



Figure 3.2: Axial composition of model core assemblies

The compositions of the assemblies used for this study represent the core at beginning of life (BOL). As a matter of fact, choosing initial assemblies composition simplifies the problem, since we attributed to the same type of assemblies the same composition, whatever are their positions.

The properties of all the elements characterizing the different materials inside the assemblies of the core, are calculated at 20 °C. We chose this temperature because of the availability of pointwise cross sections of all relevant isotopes.

20 °C is often considered as reference temperature for Monte Carlo calculations because other temperature cross sections are obtained from this point by broadening techniques.



Figure 3.3: Simplified fuel pins composition scheme

3.2 Implementation of deformed geometry in TRIPOLI- $4^{\textcircled{0}}$

3.2.1 TRIPOLI-4[®] code and settings of the simulations

The TRIPOLI-4[®] code is a three-dimensional, continuous energy code used to solve particles transport equation based on the Monte Carlo method, owned and developed by CEA.

TRIPOLI-4[®] is capable of dealing with reactor physics problems such as reproducing the behavior of particles in multiplying media, with or without fixed source, under steady state conditions.

TRIPOLI-4[®] is the latest version of TRIPOLI and it have been rewritten in more recent programming language (C and C++). It is more efficient in the description of geometrical elements (surface-based and/or combinatorial geometries) and it provides more precise representations of basic nuclear data such as the pointwise representation of cross-sections (through the use of multigroup homogenized cross sections calculated with the APOLLO3 code).^[27] In our case the code will simulate neutron transport.

We exploit TRIPOLI-4[®] power to solve time-independent neutron transport equation to calculate effective multiplication factor in several deformed core configurations in order to compare their effect on the reactivity value of the system.

For the cross sections and isotopes properties we relied on GALILEE-3 dictionary file. $^{\left[28\right]}$

To simplify and speed up the calculation, we imposed as initial distributed source only fissile and fertile volumes included in the active part of the core. The initial source is defined as isotropic and having as energy distribution, the Watt spectrum.

For the complexity and size of the geometry simulated, for our criticality calculations we set 40000 iterations with 10000 neutrons each for the convergence, considering an automatically optimized value of discarded batches necessary to
achieve equilibrium and eliminate transient initial regime.

For the simulations, it was necessary to use 60 processors and we set as maximum computational time 8 days for simpler deformed configurations and 14 days for more complex ones. More details about computational costs and convergence of the results can be found in Section 4.1.

For brevity, in the discussion of the results, we will refer to the code TRIPOLI-4[®] as simply TRIPOLI.

3.2.2 Definition of deformed core in TRIPOLI-4[®]

As inputs to TRIPOLI-4[®], we had to provide a native geometry file and a file containing the volumetric homogenized compositions of the 31 different materials that we can find inside the core (specifically they are the volumetric concentrations of all isotopes of these materials). The geometry file is the result of compiling a code which we wrote in Perl language, while the compositions file was supplied by CEA Cadarache.

TRIPOLI-4[®] enables to represent the deformation of the core as a three-dimensional exact geometry. Specifically, even though the code does not allow to model a curved volume, it is acceptable to define it as superposition of little inclined hexagonal polyhedra (see Figure 3.4). This representation results sufficiently precise on the condition that the tilt is not too relevant and that we have been using suitably small blocks.



Figure 3.4: TRIPOLI-4[®] modeling scheme seen on the axial plane

Creating the geometry code requires a complex process which can be described as a series of steps well explained in Figure 3.5.

The first step consists of the definition of the centers of the hexagons, the bases of the assemblies. They are found considering the dimension of the assemblies and the thin layer of sodium inter-assemblies. Then it is necessary to attribute



Figure 3.5: Schematic representation of steps followed for TRIPOLI native deformed geometry file

to each center the corresponding type of assembly in order to be able in the next steps to properly define their compositions, through the use of a map of the modeled core.

The second step involves the definition of each assembly. Because of the heterogeneity of the compositions of most of the types of assemblies, it is necessary to choose a proper axial mesh, that we will call "composition axial mesh", which allows us to precisely define the content of the assemblies. To choose this mesh, we took into considerations the dimensions of the layers of the materials which compose the assemblies and also we imposed to have more points in the central part of the assemblies, nevertheless ensuring comparable length of the segments. Actually two different "composition axial mesh" were defined: one of 9 points necessary to define homogeneous types of assemblies and one of 18 points for the other types. These "composition axial meshes" are needed to divide the assemblies in littler hexagonal base prisms having a certain material inside and so a certain composition.

In the third step, we are finally able to define the inclined assemblies. In order to do it, we had to create fictive volumes with the same characteristics of the ones of step 2 and then find the inclination of their rotation central axis. In order to calculate them, it is necessary to decide, what we can call, the "moving blocks mesh"; it means that we have to choose the lengths and the number of the blocks in which we want to divide the assemblies, to properly describe their deformations. In this choice, we took into account that moving blocks should have comparable sizes.

The "composition axial mesh" and the "moving blocks mesh" cannot be the

same because of the presence of very thin layers of materials in some of the assembly (of the order of few centimeters). These layers defined as volumes with a specific composition at step 2 cannot also represent moving blocks because of their small dimensions.

We decided to use 10 blocks of average 30 cm length.

Knowing that some layers of materials are much thinner than the blocks, we can imagine that in each block we can find more than one volume and material. We can see a representation of the division in blocks of the structurally more complex type of assembly in Figure 3.6.



Inner core fuel assembly

Figure 3.6: Inner core fuel assembly divided into mobile blocks

The utility of having defined the "moving blocks mesh" is that the deformation file, where we can find the displacements along x and along y of the center of the hexagons, provides them at the axial points identified by the "moving blocks mesh". Considering the new positions of the extremes of the blocks, we find the different inclinations to use for our blocks rotation. If more than one volume is comprised in the same block, it means that these volumes will have the same inclination of central rotation axis, which is also the same of the block they belong. It is important to underline that, in this case, the compositions are not homogenized inside the single moving block.

It is necessary to focus our attention on the fact that defining a continuous curved geometric volume as superposition of smaller polyhedra implies the appearance of artificial spaces, which are going to be replaced by sodium interassemblies from one side, and the overlapping of two blocks to the other. The first effect has the consequence to increase the absorption rate because of the presence of sodium rather than fissile or fertile material, while the second effect is going to reduce the available fissile or fertile zone, in the case we were examining blocks of the fuel assemblies (the more interesting one for this reflection). This two effects can combine together resulting in a reduced value of reactivity with respect to the exact geometry one. To mitigate this problem in the definition of the deformed geometry, we exploited the solution explained in Figure 3.7.



Figure 3.7: TRIPOLI-4[®] moving blocks modeling representation

We numerated the blocks from 1 to 10 and we defined the even ones as lengthened of a very small segment $2 \cdot \delta l$, in order to cover the space which is formed at the convergence of two blocks. Then we impose that odd volumes override the even ones. The length of δl was experimentally chosen and depends on the deformation of the assemblies and so, on the inclination of the blocks, but it is of the order of magnitude of thousandths of a centimeter.

Regarding as the definition of the rest of the core in the geometry file, we created a big hexagonal polyhedron of about 2.75 m of base radius and 3.10 m high, filled by sodium, where all the assemblies are immersed. For TRIPOLI calculations we supposed void boundary conditions around the 15 ring core; it means imposing null incoming flux of particles at the geometry limits in solving transport equation. In order to simulate void boundary conditions, we design this big polyhedron to intersect the outer-most ring of assemblies in the radial plane and to cut the excess of length of the assemblies, which are slightly lengthened because of the curvature (see Figures 3.8 and 3.9). After having defined another bigger hexagonal base box around the last one, we impose to fill the space between the two with an additional material, a gas with very low concentration, which enables to reproduce the void (see Figures 3.10 and 3.11).



Figure 3.8: View of the simulated geometry from above



Figure 3.9: View of the simulated geometry from the central axial plane



Figure 3.10: Detailed view of the external boundary from above



Figure 3.11: Detailed view of the external boundary from the central axial plane

3.3 Implementation of deformed geometry in Cast3m

3.3.1 Mobile mesh method on Cast3m

Mobile mesh method represents a different way to treat the deformation of the core with respect to Monte Carlo codes. It is a method which enables to directly calculate the reactivity effect of structural modification of a reactor core with a deterministic code (directly means not exploiting perturbation theory and valid not only for small deformations). This method provides the possibility to deform the geometry and the mesh together in the same environment. Therefore neutronic-mechanical coupling is facilitated but as drawback it results impossible to understand if the impact on neutronics is due to mechanical displacements or it has to be attributed to numerical effect hided in the modification of the mesh. In order to use this approach, some simplifications in mesh and neutronic parameters handling have been considered in this method.

Implementing mobile mesh method on the mechanical code Cast3m^[29] was accomplished by C. Patricot during his PhD thesis through the development of a neutronic solver CNTT (Cast3m Neutron Transport Tool).

Cast3m was chosen because it already owns algorithms solving differential equations with finite elements methods and moreover it allows to easily move the computing nodes.

For our steady state criticality calculations, CNTT solves the multigroup steady state diffusion equation with Dirichlet boundary conditions, using finite elements methods to simplify the equation and power iterations method to converge to the solution.^[9]

3.3.2 Definition of deformed core in Cast3m

In order to apply CNTT, implemented in Cast3m, to deformed geometries, it is necessary to figure it out how to describe them within the code.

Initially it was thought to provide two different size of mesh to define changes in the structure of the geometry: one to be used for the assemblies and one for the layers of sodium between the assemblies, as shown in Figure 3.12. Having small size cells which undergo big deformations could cause numerical errors, though.

Moreover, mesh deformation in Cast3m is performed translating the nodes of the cells in the horizontal plane; it leads to a description of the axial deformation of the assembly as it is showed Figure 3.13. It provides conservation of the volume of the assemblies (and so conservation of the fissile mass contained in them) but not of the exchange area between the assembly and the adjacent inter-assemblies sodium. As a matter of fact, as shown in Figure 3.13, the modeling of the deformation produces an enlargement of the lateral area of the assembly which results in an increase of moderation rate operated by the sodium and consequently in a artificial decrease of reactivity.

In order to avoid this problem, it seemed convenient to provide a definition of the mesh with homogenized cells which include the assembly but also an half of the layer of sodium which separate it from the adjacent assemblies. This second approach can be visualized in Figure 3.14. In this case, it is necessary to face the problems of handling the displacements of the assemblies and of the mass balance that has to be conserved. Each node of the mesh is displaced of



Figure 3.12: Cast3m mesh of assemblies and inter-assemblies sodium. Left : nominal configuration. Right : deformed configuration^[9]



Figure 3.13: Axial view of the deformed assembly modeled with Cast3m^[9]

the average displacement of adjacent assemblies. Specifically, the central node of the assembly is displaced on x and y directions of the value contained in a deformation file similar to the one used for TRIPOLI-4[®], while external nodes of the cells composing the assembly are placed on the gravity centers of triangles, formed by central nodes of adjacent assemblies.

In Figure 3.14, it is possible to see how mesh changes in the case of compaction of assemblies (the three on the top right).

The homogenized mesh deformation does not preserve the mass balance and for this reason, it is necessary to modify the cross sections. Particularly, cross sections are parametrized on the inverse of the surface of a given cell. The idea we want to exploit for this parametrization, is that if the cell reduces its size with the deformation, also inter-assemblies sodium amount is lower in the homogenized cell and viceversa.

We needed 33 energy groups macroscopic cross sections of all the materials (provided by staff from CEA Cadarache) calculated in self-shielding modality, homogenizing the materials with different volume fractions of inter-assemblies sodium: no inter-assemblies sodium in the cell, nominal layer of inter-assemblies sodium in the cell, double layer of inter-assemblies sodium in the cell (see Figure 3.15). The effect of this process simulates the effect of reducing and enlarging of the cell of a certain material.

The Cast3m solver, solving the diffusion equation for the single cell, uses homog-



Figure 3.14: Example of homogenized mesh deformation in Cast3m^[9]

enized macroscopic cross sections interpolated by the size of the cell between the ones we introduced before.

Also for the other parameters appearing in the diffusion equation, fission spectrum and diffusion coefficient (defined as function of total macroscopic cross section), we proceeded in the same way.

In [9], reactivity effects due to static gradual flowerings of Phénix core, obtained applying the Cast3m solver, have been validated comparing them with the ones computed by TRIPOLI-4[®] and APOLLO3 (the deterministic code used by CEA)^[30].

Moreover, [23] regards the validation of CNTT in evaluating reactivity effects due to some considered mechanical scenarios on SFRs.

As in TRIPOLI-4[®], in Cast3m it is not possible to define curved volumes. To reproduce this effect, we divided the assembly in inclined blocks having the center of the base shifted of given δx and δy on x and y axes.

Concerning the axial mesh, meaning the choice of the axial length of the blocks we used to define the axial deformation of the assemblies, we chose the lengths of the blocks maintaining them comparable to more efficiently reproduce the deformation. These blocks were not necessarily homogeneous in composition so if the block contains more than one material, macroscopic cross sections were averaged on the volume occupied by each material, that in this case (with size of the base fixed for the single block), means averaging on the height of each layer with respect to total height of the block. This simplification introduced an additional source of numerical error in the calculation and for this reason it seemed appropriate to use a higher number of blocks to define the assembly in order to reach a good accuracy. As a matter of fact, we performed two calculations for each deformed configuration of the core: one using 14 blocks, the other 22 blocks to define the axial deformation of the assemblies.

In Figure 3.16 it is possible to visualize how Cast3m reproduces the deformation of the geometry: the blocks are inclined having the coordinates of the centers



Figure 3.15: Different cells homogeneization with different inter-assemblies sodium fractions

of the bases modified by the δx and δy , contained in the deformation file at each axial mesh point, and their compositions is homogenized considering the composing materials with their volume fractions.

It is not easy to implement the void boundary conditions in CNTT, solving diffusion equation, so we decided to simply set zero flux boundary condition in this case.

For brevity, in Chapter 4, discussing the results, we will refer to calculations performed considering 15-point axial mesh as Cast3m15 and the ones performed considering 23-point axial mesh as Cast3m23.



Figure 3.16: Definition of deformed geometry in Cast3m

3.4 Specifications of selected deformed arrangements of the core

For this analysis, we chose different configurations of deformed core, in order to be able to identify some trends in the reactivity effects.

In this section, we describe all the configurations on which the criticality calculations with the two code were performed: some of them represent little increasing deformations of the core, while others reproduce relevant deformations caused for example by particular accidental sequences.

3.4.1 Straight compactions and widenings of the core

Firstly, we considered straight translations of the assemblies of the core in outward and inward radial directions.

When the translation is directed towards the central assembly, which remains in the same original position, we are reproducing a compaction of the core (see Figure 3.17). In this case the layer of inter-assemblies sodium is thinner than in the nominal configuration.

We performed four calculations with different levels of compaction. We can identify these levels calculating how much the layer of inter-assemblies sodium is reduced with respect to the nominal value.

The names we used for the different levels are related to the names of corresponding deformation files provided by CEA Cadarache and they are useful to be identified by them.



Table 3.1:	Definition	of the	levels	of	compaction
------------	------------	--------	--------	----	------------

Name	Level of compaction	Change of inter-assemblies Na layers
Comp2	1	- 6.6 %
Comp4	3	- 19.8 %
Comp8	7	- 46.2 %
Comp14	13	- 85.8 %

In Table 3.1, it is possible to see how the levels of compactions are defined as function of the change in the thickness of inter-assemblies sodium layer: level one corresponds to a reduction of it of 6.6 %, level three to the triple of this value. Viceversa, if the translations are directed radially towards the outer part of the core, we are talking about widenings of the core (see Figure 3.18).

As for the compactions, we performed four calculations corresponding to four increasing levels of widening and they are defined in relation to the increase of the thickness of the layer of sodium between the assemblies (see Table 3.2).



Figure 3.18: Visualization of core widening

In these cases, the assemblies remain aligned and, for this reason, the length δl of which the even number blocks are elongated in TRIPOLI to avoid sodium to fill the openings formed between two inclined blocks (see Subsection 3.2.2), is set to 0.

Name	Level of widening	Change of inter-assemblies Na layers
Gerb2	1	+ 6.6 %
Gerb4	3	+ 19.8 %
Gerb8	7	+ 46.2 %
Gerb14	13	+ 85.8 %

Table 3.2: Definition of the levels of widening

3.4.2 Deformation of the core due to a 6MJ explosion

In order to test the feasibility of the descriptions of an heterogeneous core deformed in TRIPOLI-4[®] and Cast3m as the ones we defined in Subsections 3.2.2 and 3.3.2, the deformations which undergo this core at some timesteps of a transient reproducing the sequel of a 6MJ explosion at its top center is carried out. We performed static criticality calculations on the resulting deformations of this accidental transient, registered at the timesteps showed in Table 3.3.

Table 3.3: Timesteps of the post-explosion transient at which correspond our static criticality calculations

Name	Timesteps of the transient
6MJ_1	0.1 s
6MJ_2	0.2 s
6MJ_3	0.3 s
6MJ_4	0.4 s
6MJ_maxcomp	$0.456 \mathrm{~s}$
6MJ_5	$0.5 \mathrm{~s}$
6MJ_6	0.6 s
6MJ_7	0.7 s
6MJ_8	0.8 s
6MJ_9	0.9 s
6MJ_10	1.0 s

Structurally this transient represents firstly a flowering of the core right after the explosion, followed by a "rebound" effect resulting in a compaction of the core which reaches its maximum at 0.456 s (timestep of maximum compaction). The expected variation of reactivity (defined as the difference between the reactivity measured on the deformed configuration and the one computed for the original geometry) versus time behaviour should show firstly a decrease, then an increase reaching its maximum at 0.456 s and it should finally stabilize towards zero.

These 11 deformation files were provided by CEA Cadarache in the format: identification number of the assembly, height at which the deformation corresponds, deformation on x axis with respect to the original central point (δx), deformation on y axis with respect to the original central point (δy).

The axial points at which we received the information about the deformation by staff from CEA Cadarache were not the same chosen by our meshes, the one used for TRIPOLI made of 11 points and the two used with Cast3m of 15 and 23 points. In order to compute them, we linearly interpolated between the data we had by CEA Cadarache, introducing clearly a source of error in the representation of the real geometry, which can be not negligible in this case where the deformation of the core is important.

About the length δl needed to be set for a more correct definition of the deformed assemblies in TRIPOLI, it was experimentally verified that it has to take the value $\delta l=0.005$ cm.

3.4.3 Flowerings of the core starting both from the central assembly and from a more external one

Since, as we argued at Subsection 2.2.2, core flowering in Sodium-cooled Fast Reactors is considered of relevant interest, we decided to test how reactivity effects computed with Cast3m ensues in comparison with the reference variation of reactivity ($\Delta \rho$) calculated by TRIPOLI-4[®].

For core flowering is intended when assemblies "flower" around the central one: this means that they stay in the same original position at the base but they are inclined on the axial plane, progressively axially increasing the thickness of sodium layers between the assemblies.

On the top view they progressively spread out in radially outward directions, with increasing axial height starting from the lower bases. This configuration reproduces a geometry where assemblies are closer the one to the other in the lower part than in the upper part of the core. In Figure 3.19 this type of deformation is visualized.



Figure 3.19: Visualization of core flowering from the center

Name	Levels of flowering	Inclination angle of the assemblies [°]
Flower_0.5	1	0.09
Flower_1	2	0.18
Flower_1.5	3	0.27
Flower_3	6	0.54
Flower_5	10	0.9

Table 3.4: Definition of levels of flowering around central assembly

The deformation files for the core flowering test were prepared setting an inclination of the axes of the assemblies of angles defined in Table 3.4 in radial outward directions.

Imposing the flowering of the geometry centered around a certain location means imposing in this location an increase of the concentration of sodium.

In order to see how the location of the center of the flowering influences the reactivity effects due to the deformation, we performed some radial flowering calculations but, in this case, not around the central assembly but around an assembly located on the bisector of the top-right quadrant on the fifth ring of the core, as you can see in Figure 3.20.



Figure 3.20: Visualization of core flowering around an assembly belonging to the fifth ring

For reason of time needed by the calculations to be completed, we only considered three tilting angles of the axes of assemblies in outward radial directions, for this kind of configurations (see Table 3.5).

Table 3.5: Definition of levels of flowering around an assembly belonging to the fifth ring

Name	Levels of flowering	Inclination angle of the assemblies $[\circ]$
Flower_1_66	1	0.18
Flower_3_66	3	0.54
Flower_5_66	5	0.9

For these types of deformations, the single assembly remains aligned and so there is no reason to set δ l different from 0.

Chapter 4

Discussion of the results

In this chapter we present the results involving the reactivity values computed with the reference Monte Carlo code for each deformed configuration considered and the comparison of reactivity effects due to the deformations calculated with TRIPOLI-4[®] and Cast3m.

In the first part we point out some relevant information regarding the convergence criteria used and related computational cost of the different simulations. In particular, we specify the reasons which forced us to make some choices about the axial mesh or about the number of configurations analyzed.

In the second part we show graphs concerning the results, computed with the two codes, and we comment the relation between the effect and the type of deformation trying to argue the motivations underneath.

4.1 Computational costs and convergence criteria of calculations of TRIPOLI-4[®]

In the output file of TRIPOLI-4[®] we find the value of k-effective of the system with its relative standard deviation.

The effective multiplication factor, named k-effective (k_{eff}) , is defined as the ratio between the number of neutrons produced by fission and the number of neutrons lost by absorption and leakage at a certain generation.

In order to take into account the change in k_{eff} with respect to critical state, another quantity is defined, reactivity ρ , which is defined in function of k_{eff} as:

$$\rho = \frac{k_{eff} - 1}{k_{eff}}.\tag{4.1}$$

Reactivity value could be positive, zero, or negative. Critical conditions corresponds to reactivity equal to zero so the larger is the absolute value of rho, the further the system is from criticality. In fact the reactivity can be seen as the measure of a reactor's relative departure from criticality.

Reactivity is a dimensionless number and, because of the quite small value it usually takes in power reactors, it is expressed in pcm, percent mille, which means one thousandth of a percent $(10^{-2} \cdot 10^{-3} = 10^{-5})$. This unit allows reactivity to be written in whole numbers.

The k-effective we used to compute the reactivity of the system was the one represented by the KCOLL estimator. It accounts for the average number of neutrons the code should produce, estimating it on the base of the collisions. The value of the standard deviation on the k_{eff} is not imposed so it is not the same for all the simulations.

Conversely we set the maximum computational time employed by the calculations in order to reach a standard deviation value lower than 6 pcm:

- 8 days for the compaction, widening and flowering deformations of the geometry;
- 14 days for the post-explosion transient deformations of the geometry.

In Appendix, it is possible to find the standard deviation reached by each simulation. It should be noticed that for some flowerings configurations, the standard deviations which we were able to obtain, were of the order of 10/15 pcm because no more time was available to let the calculations reach higher accuracy.

4.2 Computational costs and convergence criteria of calculations of the solver of Cast3m

Cast3m in its output file provides the k_{eff} (rounded to the fourth decimal place), the reactivity value of the deformed configuration and the difference between it and the nominal geometry reactivity coefficient, that we called $\Delta \rho$, expressed both in pcm.

CNTT, implemented in Cast3m, is built to reach the convergence at iteration n when the following conditions, on k_{eff} and on total neutron flux Φ with the iteration number indicated in the apexes, have been satisfied:

•
$$\left\| \Phi^{(n)} - \Phi^{(n-1)} \right\|_2 / \left\| \Phi^{(n)} \right\|_2 < 1 \cdot 10^{-5};$$
 (4.2)

•
$$\left|k_{eff}^{(n-1)} - k_{eff}^{(n-2)}\right| < 5 \cdot 10^{-6}.$$
 (4.3)

Moreover, we set the maximum number of iterations at 500, but this criterion remained always satisfied.

As we already introduced in Subsection 3.3.2, we performed calculations on the deformed geometry with Cast3m using firstly 14 but also 22 axial blocks. In fact we realized that using 14 blocks the values of k_{eff} computed with Cast3m differed of thousands of pcm with respect to the ones had with the reference TRIPOLI code. In order to reduce this gap in the multiplication factors comparison, we decided to perform the same calculations but increasing the number of points of the axial mesh from 15 to 23 (increasing in this way the accuracy in reproducing the deformation). As we can see from Table 4.1 (the first part related to the nominal geometry and the second referred to the deformation recorded at first timestep of the post-explosion transient), using 22 blocks allows to reduce the gap between the multiplication factor calculated with Cast3m and the reference one to some hundreds of pcm.

We went even further and we computed the results also using 30 points on the axial mesh(29 blocks) and this approach led to reduce even more the gap between the two values for both the configuration.

Reference case							
	TRIPOLI	Cast3m 15 points	Cast3m 23 points	Cast3m 30 points			
k_{eff}	1.06449	1.0316	1.0590	1.0609			
ρ [pcm]	6058	3066	5569	5736			
Deformation at 0.1 s after the 6MJ explosion							
	TRIPOLI	Cast3m 15 points	Cast3m 23 points	Cast3m 30 points			
k_{eff}	1.05655	1.0279	1.0545	1.0563			
ρ [pcm]	5353	2717	5169	5332			
Comparison between the deformed and nominal configurations							
	TRIPOLI	Cast3m 15 points	Cast3m 23 points	Cast3m 30 points			
$\Delta \rho [\text{pcm}]$	-705	-349	-400	-404			

Table 4.1: Convergence study to choose the number of points of Cast3m axial mesh

Nevertheless, as we can see from the last row of Table 4.1, simulating with this number of points on the axial mesh does not provide relevant improvement in the estimation of $\Delta \rho$, comparing them with the ones computed with the reference code TRIPOLI. In light of these results, we believed that performing all the deformed geometry calculations in Cast3m with 30 points would cause excessive effort in terms of computational time and cost. In fact, for our analysis, which is focused on the $\Delta \rho$ due to the deformations, the order of magnitude of the difference between the k_{eff} value computed with TRIPOLI and the one reached with Cast3m using 22 axial blocks, is acceptable.

Simpler deformed configurations needed about 6 up to 15 hours for the calculation (15 for the simulation considering higher number of axial mesh points).

Post-explosion transient deformed configurations needed about 12 to 24 hours to reach convergence, instead.

In Section 4.3 we will report the results obtained with Cast3m using both 15 and 23 axial mesh points.

4.3 Presentation of the results

In this section we will show how the reactivity value changes with respect to the nominal situation when the geometry of the core is deformed.

In order to better visualize these ρ changes, we decided to report only the values computed with the reference Monte Carlo code. The reason is that, as explained before and as you can see in the Appendix, the values of ρ obtained for the considered configurations with Cast3m differs of hundreds of pcm with respect to the reference ones and putting in comparison reactivity values was not of our interest.

For the nature of the two codes, it is obvious to expect that the Cast3m solver, relying on the diffusion theory, cannot reach the level of accuracy inherent in TRIPOLI in the calculation of the multiplication factors.

Nevertheless, the reactivity values computed with Cast3m can be found in the tables in Appendix.

We will particularly focus on the reactivity effects due to the deformations of the geometry comparing the ones calculated with TRIPOLI to the ones resulting by reproducing the deformed configurations using the two axial mesh within Cast3m.

4.3.1 Compactions and widenings

Using TRIPOLI- $4^{\textcircled{m}}$, we were able to calculate how reactivity changes with increasing levels of compaction and widening.

We are also interested in finding out if the neutronic solver of Cast3m, which has way lower computational cost and requires way less computational time, could approximate the reactivity effects induced by these two types of deformation.

As we can see from Table 4.2, increasing compaction of the core results in increasing reactivity insertion. Specifically a reduction of 6.6% of the thickness of the layers of cooling sodium between the assemblies cause an increase of reactivity of about 0.9%.

Table 4.2: Reactivity coefficient and reactivity effects of the compacted configurations computed with TRIPOLI reference code

Name	Level of compaction	ho [pcm]	$\Delta \rho \ [pcm]$	$\% \Delta \rho / \rho$
Nominal	0	6058	-	-
Comp2	1	6112	+54	+0.89%
Comp4	3	6223	+165	+2.73%
Comp8	7	6452	+394	+6.51%
Comp14	13	6796	+738	+12.19%



Figure 4.1: Reactivity changes due to compaction of the core

Comparing the definition of compaction levels in Subsection 3.4.1 with the reactivity insertion registered with TRIPOLI, we notice a linear dependence between the absolute value of the percentage change in the sodium inter-assemblies layer and the corresponding percentage change in the reactivity value, as showed in Figure 4.1.

The computed reactivity effects is, in fact, mainly due to geometrical effects on the balance between reaction rate and leakage: a compaction of the core has as consequence the increase of density of heavy materials (the ones composing the assemblies, heavier than the sodium) inside the core, which causes the rise of the probability of interaction of particles (and consequent decrease in leakage probability). Moreover, if as compaction effect, the amount of sodium inter-assemblies is reduced, the neutron spectrum is hardened and consequently fission rate increases and at the same time the absorption rate by sodium is reduced. These combined effects lead to the increase of the reactivity coefficient. The linear proportionality between the reactivity insertions and the compaction levels we observed, reminds us the possibility of using the perturbation theory to calculate the effect of this type of deformation, as we discussed previously in Chapter $3.[^{31}]$

In Figure 4.2 it is possible to see how Cast3m simulations with 15 and 23 points of the axial mesh approximate the reactivity effects calculated with the reference code.

The reactivity variation, $\Delta \rho$, is slightly underestimated by Cast3m.



Figure 4.2: Reactivity insertion due to compaction of the core

As we could imagine, using more points on the axial mesh improves the estimation of the $\Delta \rho$ by the code with respect to the reference one computed by TRIPOLI, going from the 10/15% to 3/6% of error.

The error in the estimation of the reactivity effect by Cast3m increases for higher

levels of compactions (see Table 4.3).

Table 4.3: Comparison between reactivity effects computed with TRIPOLI and Cast3m (compactions)

Name	Compaction	$\Delta \rho \ [pcm]$	$\Delta \rho \ [pcm]$	$\Delta \rho \ [pcm]$	% Error $\Delta \rho$	% Error $\Delta \rho$
	levels		Castem 15	Castem 23	Cast 3m 15	Cast 3m 23
Comp2	1	+54	+49	+53	-10.1%	-2.7%
Comp4	3	+165	+146	+158	-11.8%	-4.3%
Comp8	7	+394	+340	+370	-13.7%	-6.2%
Comp14	13	+738	+634	+690	-14.2%	-6.6%

The same approach was used for the analysis of the widening configurations. The reactivity values computed with the reference Monte Carlo code for the different widening configurations can be found in Table 4.4.

Table 4.4: Reactivity coefficient and reactivity effects of the widening configurations computed with TRIPOLI reference code

Name	Level of widening	ρ [pcm]	$\Delta \rho \ [pcm]$	$\% \Delta \rho / \rho$
Nominal	0	6058	-	-
Gerb2	1	5998	-60	+0.99%
Gerb4	3	5884	-174	+2.87%
Gerb8	7	5655	-403	+6.65%
Gerb14	13	5321	-737	+12.16%

For the opposite reasons we argued before about compactions, we could expect that increasing levels of widening of the core would correspond to increasing reduction of the reactivity coefficient.

We can notice that increasing the layer of sodium inter-assemblies of 6.6% produces 1% reduction of the reactivity value; the effects of the first 3 levels of widening result slightly higher than the ones computed for the opposite compacted configurations.

In Figure 4.3 we can observe, also in this case, the linearity between the levels of widening and the corresponding reactivity variations computed with TRIPOLI. Regarding the convergence of reactivity effect values computed with Cast3m to the ones had by TRIPOLI we can observe that in these cases, Cast3m underestimates the change in reactivity produced by the widenings and this underestimation is higher at low level of widening (see Figure 4.4).

The error in the estimation by Cast3m is higher than in the previous case dealing with compactions. Infact, it attains between 13/17% using 15 axial mesh points and 6/10% using 23 axial mesh points.



Figure 4.3: Reactivity changes due to widening of the core

Table 4.5: Con Cast3m (wider	nparison betv nings)	veen reactivity	effects compu	ted with TRIPO	LI and
Widening		$\Delta \rho \ [pcm]$	$\Delta \rho \ [pcm]$	% Error $\Delta \rho$	% Error

Name	Widening levels	$\Delta \rho \ [pcm]$	$\Delta ho ~ [pcm] \ Castem 15$	$\Delta ho~[pcm] \ Castem 23$	$\% \ Error \ \Delta ho$ Cast $3m15$	$\% \ Error \ \Delta ho$ Cast3m23
Gerb2	1	-60	-50	-54	+16.7%	+10.0%
Gerb4	3	-174	-148	-161	+14.9%	+7.5%
Gerb8	7	-403	-346	-374	+14.1%	+7.2%
Gerb14	13	-737	-641	-693	+13.0%	+6.0%

We can conclude that the Cast3m solver capacity in reproducing the reactivity effects of deformed configurations is improved when we are dealing with a compaction of the core rather than a widening of the same size. This result could be due to uncertainties in the calculation of homogenized cross sections of the cells when the concentration of sodium inside increases (corresponding in widening of the core). We will treat better this problem in Subsection 4.3.2, related to post-explosion transient deformations where the layers of sodium between the assemblies modify significantly their thickness.

In Figure 4.5, we plotted together the behaviour of $\Delta \rho$ for symmetric levels of compaction and widening and we can visualize the symmetric trends of red and green lines (representing $\Delta \rho$ calculated with Cast3m using 15 and 23 blocks). Therefore Cast3m, in absolute terms, provides a slight underestimation of the

reactivity effects due to these simple types of deformation.

Finally, we can state that, even though Cast3m was not so efficient in providing k_{eff} values comparable to the ones obtained with the reference code, it turned out to be very reliable in estimating the values of the reactivity effects induced by the simple types of deformation we deal with in this paragraph.



Figure 4.4: Reactivity insertion due to widening of the core



Figure 4.5: Comparison of reactivity insertion due to same levels of compactionwidening

4.3.2 Deformations during the transient following a 6MJ explosion on the top of the core

The deformations of the transient resulting after a 6MJ explosion on the top center of the core are more complex of the ones considered before because, in this case, blocks on the axial plane are not aligned but they have different inclinations.

As we said before, the deformation files describing the deformation of the geometry at some timestep of the 1 second transient have not any regularity: it means that each assembly moves independently from the others and all we can observe is the global modification of the geometry of the core.

This transient reproduces a flowering for the first recorded timesteps and then a recompaction which passes through timestep t=0.456 s when the maximum compaction of the core is registered. Then the nominal configuration is almost restored after 1 second from the start of the transient.

In Table 4.6, we can see that initially the flowering produces a negative reactivity insertion up to 14% with respect to the nominal situation.

Name	Timestep of the transient [s]	ρ [pcm]	$\Delta ho \ [pcm]$	$\% \Delta \rho / \rho$
Nominal	0	6058	-	-
6MJ_1	0.1	5353	-705	-11.6%
6MJ_2	0.2	5211	-847	-14.0%
6MJ_3	0.3	5729	-329	-5.4%
6MJ_4	0.4	6208	+150	+2.5%
6MJ_maxcomp	0.456	6265	+207	+3.4%
6MJ_5	0.5	6263	+205	+3.4%
6MJ_6	0.6	5957	-101	-1.7%
6MJ_7	0.7	5739	-319	-5.3%
6MJ_8	0.8	5696	-362	-6.0%
6MJ_9	0.9	5816	-242	-4.0%
6MJ_10	1.0	5959	-99	-1.6%

Table 4.6: Reactivity coefficient and reactivity effects due to deformations of the transient following a 6MJ explosion on top of the core, computed with TRIPOLI reference code

As we could expect from the discussion in 4.3.1, the "rebound" effect resulting in the compaction of the core brings a maximum positive reactivity insertion of about 3.5% recorded at timestep corresponding to the maximum compacted configuration of the geometry. At the end of the considered transient, the reactivity value differs of about 100 pcm with respect to the nominal configuration value. In Figure 4.6, we report a graph of the reactivity coefficients calculated for each deformed configuration with the Monte Carlo reference code as function of time passed from the explosion event.

As regard the performance of Cast3m in estimating the reactivity effects caused by this complex type of deformation, we can notice from Table 4.7 that the neutronic solver based on diffusion theory is not behaving very well for the initial flowering of the core right after the explosion and also in occasion of the reflowering of the core after t=0.5 s.



Figure 4.6: Reactivity changes due to 6MJ explosion on the top of the core

Table 4.7: Comparison between reactivity effects computed with TRIPOLI and Cast3m (deformations of the transient following a 6MJ explosion on top of the core)

Name	Timestep of the transient [s]	$\Delta \rho \ [pcm]$	$egin{array}{lll} \Delta ho ~ [pcm] \ Cast 3m15 \end{array}$	$egin{array}{lll} \Delta ho ~ [pcm] \ Cast 3m23 \end{array}$	$\% \ Error \ \Delta ho$ Cast 3m15	$\% \ Error \ \Delta ho$ Cast $3m23$
6MJ_1	0.1	-705	-349	-400	+50.5%	+43.3%
6MJ_2	0.2	-847	-462	-517	+45.5%	+39.0%
6MJ_3	0.3	-329	-128	-166	+61.1%	+49.5%
6MJ_4	0.4	+150	+149	+139	-0.3%	-7.3%
6MJ_maxcomp	0.456	+207	+170	+192	-17.9%	-7.2%
6MJ_5	0.5	+205	+187	+191	-8.8%	-6.8%
6MJ_6	0.6	-101	-26	-63	+74.3%	+37.6%
6MJ_7	0.7	-319	-154	-199	+51.7%	+37.6%
6MJ_8	0.8	-362	-176	-222	+51.4%	+38.7%
6MJ_9	0.9	-242	-107	-146	+55.8%	+39.7%
6MJ_10	1.0	-99	-20	-47	+79.8%	+52.5%

Graphically, from Figure 4.7, we can see how the gaps between the blue and the other two lines significantly grow when the deformed configurations deviate substantially from the original one. In these cases Cast3m overestimates $\Delta \rho$ and the percentage error in the estimation is of the order of 50 to 80% if we consider 14 blocks, of the order of 35 to 50% considering 22 blocks to describe the assemblies modifications. When we are dealing with the compaction of the core, between 0.4 and 0.5 seconds after the explosion, Cast3m solver is working way better than for the other timesteps, instead.



Figure 4.7: Reactivity insertion due to 6MJ explosion on the top of the core

Looking back to 3.3.2, where we described the definition of the deformed geometry in Cast3m, we could understand the reasons beneath this difference we underlined: the problem of representing a core undergoing flowering in Cast3m could be linked to the way homogenized macroscopic cross sections are calculated for the Cast3m cells. In fact, we calculated the macroscopic cross sections, to use as inputs fo Cast3m, interpolating between 3 different configurations of the cells considering 3 levels of concentration of sodium inside the cells. Probably, the reason of the problem showed by Cast3m is hided behind not accurate calculation of the macroscopic cross sections used for the calculations. If the flowering causes locally accumulations of sodium between the assemblies of more than the double of the nominal layer size, it would lead to wrong derivation of the cross sections for Cast3m.

Moreover, we are encouraged to think that wrong cross sections could have led to have such an high error in the estimation of the $\Delta \rho$ because of the good performance of the code in describing the compactions which undergo the core at timesteps 0.4 s, 0.456 s, 0.5s.

As regards compactions, in fact, the derivation of the macroscopic cross sections could not be a source of error.

For these compactions, Cast3m is able to approach the reactivity effects calculated with TRIPOLI-4[®] pretty accurately (estimation error of the order of 7/8%

using 23 axial mesh points).

Another result which deserves to be highlighted is that, differently with respect to what we found in Subsection 4.3.1, higher number of blocks used for the description of deformed assemblies is not always guarantee of a better estimation of $\Delta \rho$; for example at t=0.4 s a lower number of blocks has provided a better result than the one obtained using more blocks.

We could state that an increasing number of blocks for the description of the deformed geometry in Cast3m allows to get closer to the k_{eff} values calculated with TRIPOLI but it is not assurance of better estimation of $\Delta \rho$. This is due to the fact that, when we are dealing with irregular deformation of the assemblies, also the definition of the length of the blocks influence the correct description of the deformed assemblies; this is especially true for Cast3m because of the mesh deformation method considering homogenized cells.

We could explain what happens for the estimation of $\Delta \rho$ at timestep 0.4 s thinking that probably the choice of the length of the 14 blocks would fit better the actual deformation which undergo the assemblies which allows them to describe the deformed configuration more accurately.

4.3.3 Flowerings starting from the center of the core and from a more external assembly

In order to test the two codes for some flowering configurations, we chose to simulate deformed cores made of assemblies having inclined central axis of certain increasing angles in radial outward directions, around a central one, which stays fixed.

The sizes of the flowerings, we chose, provide displacements of the assemblies of the order of centimeters on the top. This displacement cause relevant increase of the layers of sodium between the assembly considered as the source of the flowering and the adjacent ones, especially on the top of the core. Despite of the relevant modification of the core structure caused by flowerings of this size, for the way we defined the deformed geometry in TRIPOLI, we know that the Monte Carlo code is able to exactly reproduce this type of deformations.

From Table 4.8 we can see the reactivity values computed with TRIPOLI for flowering of the core around the central assembly.

Name	Widening levels	ρ [pcm]	$\Delta \rho \ [pcm]$	$\% \Delta \rho / \rho$
Nominal	0	6058	-	-
Flower_0.5	1	5941	-117	-1.9%
Flower_1	2	5838	-220	-3.6%
Flower_1.5	3	5733	-325	-5.4%
Flower_3	6	5440	-618	-10.2%
Flower_5	10	5086	-972	-16.0%

Table 4.8: Reactivity effects due to increasing flowerings around the central assembly

Since the flowering represents a local decrease in density of heavy materials and a consequent increase of local concentration of sodium coolant around to the position which works as the source of the expansion, we would expect negative insertion of reactivity.

Higher levels of flowering correspond to higher reactivity effects but they are not exactly linearly proportional to the levels of flowering.

As shown in Table 4.8 the relation between $\Delta \rho$ computed with TRIPOLI -4[®] and the levels of flowering are not exactly linearly proportional. The effect in reactivity is proportionally higher for small flowerings than for larger ones. This result, different from what we obtained for widening translations in Subsection 4.3.1 should be due to the geometrical effects of these expansions, which do not affect all the core uniformly, but it has higher impact on the upper part of the core than on the lower one.



Figure 4.8: Reactivity changes for core flowering starting from the center

As regard Cast3m performance in computing the reactivity effects due to this type of deformations, we could already say that for the higher levels of flowering the code has the same problem explained in Subsection 4.3.2, such as wrong derivation of macroscopic cross sections used in Cast3m because of excessive concentration of inter-assemblies sodium in the deformed cells. In fact for small size flowerings (levels 1 and 2) Cast3m estimates the $\Delta \rho$ computed with the reference code with a percentage error of the order of 30% (using 14 blocks for the assembly axial definition), 15% (using 22 blocks).

Nevertheless, also in these cases, the trend showed by Table 4.9 catches Cast3m overestimating the $\Delta \rho$ values just because it underestimates the increase of concentration of sodium inter-assemblies for the well explained problem with the cross sections calculation.

For levels of flowering higher than 3, Cast3m is no more reliable. In fact the corresponding $\Delta \rho$ values computed with Cast3m are not going to increase with

the increasing of flowering levels, as we can see graphically in Figure 4.9 where the red and green lines flatten between flowering level 3 and 10.

Widening $\Delta \rho \ [pcm]$ % Error $\Delta \rho$ % Error $\Delta \rho$ $\Delta \rho \ [pcm]$ $\Delta \rho \ [pcm]$ Name level Cast3m15Cast3m23 Cast3m15Cast3m23Flower 0.5 -117 -103+29.1%+12.0%1 -83 Flower 1 2 -220 -155-188 +29.5%+14.5%Flower 1.5 3 -325 -190 -226 +41.5%+30.5%Flower 3 6 -618 -231 -269 +62.6%+56.5%+71.5%Flower 5 10 -972 -244 -277+74.9%



 Table 4.9: Comparison between reactivity effects computed with TRIPOLI and

 Cast3m (flowering around the central assembly)

Figure 4.9: Reactivity insertion for core flowering starting from the center

Maintaining the definition of the flowerings as before, we decided to perform criticality calculations also on geometry undergoing flowerings centered not in the center of the core, but in the position of a fifth ring assembly. In this position, the impact on the balance between reaction and leakage rates is different than in the previous case.

We could expect that for the same level of flowering the reactivity effect would be lower comparing it to the one registered when the flowering is centered around the central assembly. This is due to the different importance of the two regions, where the decrease of density and accumulation of sodium occurs, in terms of neutron flux. In fact, usually neutron flux reaches its maximum close to the center of the core and then decreases going towards the boundary.

An increase of absorption rate at about 1 meter radially from the center of the core produces a lower effect with respect to the same increase localized around the central assembly.



Figure 4.10: Reactivity changes for core flowerings around the central assembly and around a fifth ring assembly

Table 4.10: Reactivity changes computed with TRIPOLI in case of flowerings around a fifth ring assembly

Name	Widening level	ρ [pcm]	$\Delta \rho \ [pcm]$	$\% \Delta \rho / \rho$
Nominal	0	6058	-	-
Flower_1_66	2	5900	-158	-2.6%
Flower_3_66	6	5577	-481	-7.9%
Flower_5_66	10	5258	-800	-13.2%

In Figure 4.10, it is possible to visualize this difference : we reported in the same graph the reactivity changes (computed with TRIPOLI) due to the same flowering levels when the flowering is around the center of the core and when it is localized around a more external assembly.

The black line, as we already supposed, results less sharp than the blue one. As shown in Table 4.10 and in Figure 4.10, the linearity between the reactivity effects (calculated with TRIPOLI) and the levels of flowering is restored for this type of flowerings.

Concerning Cast3m results in this case, since it was not possible to perform calculations for the all the levels we test before, we are able only to say that Cast3m estimates pretty accurately the $\Delta \rho$ for flowering level 2 (absolute value of the percentage error going from 7 to 9% depending on the number of blocks used fro the description).

As showed in Figure 4.11, for higher levels of flowering, Cast3m is no more reliable in computing the reactivity effects, for the same reason explained before.



Figure 4.11: Reactivity insertion for core flowering with the source on a fifth ring assembly

Table 4.11: Comparison between reactivity effects computed with TRIPOLI and Cast3m (flowering around a fifth ring assembly)

Name	Widening level	$\Delta \rho \ [pcm]$	$egin{array}{llllllllllllllllllllllllllllllllllll$	$\Delta ho [pcm] \ Cast 3m 23$	$egin{array}{c} \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \$	$egin{array}{c} \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \ \$
Flower_1_66	2	-158	-144	-170	+9.4%	-6.9%
Flower_3_66	6	-481	-210	-242	+56.3%	+49.7%
Flower_5_66	10	-800	-214	-245	+73.3%	+69.4

Chapter 5

Conclusion and suggestion for future work

The aim of this work is to analyze different deformed configurations of a Sodiumcooled Fast reactor core.

The particularity, which makes this work unique, is the core, object of our simulations: not only radially but also axially heterogeneous core.

The fact that we had to deal with deformations of assemblies which have not an axial homogeneous composition provides a lot of effort in the actual definition of the deformed assemblies in the two codes we used, TRIPOLI-4[®] and Cast3m. To provide a precise 3D description of the deformations of the assemblies, also considering their axial composition, we defined them as superposition of hexagonal base polyhedra composed by one or more materials, whose axes are rotated of an angle, that is function of the displacements of the centers of the two bases. This is what we implemented in a Perl code to produce the geometry to use as input for the Monte Carlo code developed by CEA, TRIPOLI-4[®].

We considered the reactivity values calculated by TRIPOLI as the reference results for the ones computed with a deterministic code, a neutronic solver based on diffusion theory, developed by C. Patricot in the Cast3m environment in order to realize mechanical-neutronic coupling. The capabilities of this code were already tested on other systems (as in [9], [22], [23]), but in this work we used it to reproduce very complex deformations of axially heterogeneous assemblies. We know that, in conditions when reactivity changes fastly due to structural deformation of the core in case of accidental conditions, such as what we described in Subsection 3.4.2, having a tool which is able to estimate the consequent positive or negative insertion of reactivity with a good accuracy and low computational cost is fundamental.

In this analysis, we took into account different deformed configurations, from the simpler compactions and widenings to the complex deformations, which undergo a SFR core during a transient following a 6MJ explosion on the top.

We demonstrated that the effects in reactivity we expected for these configurations were effectively recorded by the criticality calculations we performed with the reference Monte Carlo code.

In fact, for increasing levels of compaction involving all the assemblies of the core, TRIPOLI registered a proportional increasing of the values of the reactiv-

ity insertion.

The same we can say about the increasing levels of widening that, conversely, provided increasing insertions of negative reactivity in the system.

We observed a linear dependence between the levels of compaction or widening and the effects on the reactivity.

We explained this result linking the levels of compaction to the physical impact these compactions has on the core: they provide the global increase of density of the core because the materials, which compose the assemblies that are getting closer, are heavier than sodium they are displacing. This has as result an increase in the interaction rates: fission and absorption rates.

At the same time the thinning of the layers of inter-assemblies sodium causes lower moderation of neutrons by the coolant sodium and so, the increase of fission rate due to the hardened spectrum and the reduction of absorption rate due to the decrease of volumetric concentration of sodium inside the core, with respect to the nominal geometry.

The same reasoning can be done with opposite effects when we are talking about widenings of the core.

When we decided to analyze the deformations expected during a transient following a 6MJ explosive event occurring on the top of the core, we already knew that, because of the irregularity of the 11 configurations we took into account, it would be difficult to predict exactly the trend of the reactivity effects of the deformations.

We knew that the modifications of the structure corresponding to the first timesteps of the post-explosion transient would reproduce a flowering of the core, followed by a compaction registering its maximum at timestep t=0.456 s, and that then the initial reactivity value would be almost restored.

Effectively the results in terms of $\Delta \rho$, calculated as the difference between final reactivity value and the nominal one, have been congruent with the characteristics of the simulated configurations.

We also considered another simple type of deformations, which does not consist this time in straight assemblies widened but in the inclination of a certain angle in the radial outward directions of all the assemblies composing the geometry around a central fixed one (flowering).

This deformation maintains the alignment of the assemblies providing at the same time the axial displacements of the blocks defining the assemblies.

We took in consideration configurations reproducing some levels of flowering around the central assembly and also some reproducing the same levels of flowering around an assembly located in the fifth ring, distant about 1 m from the center, in order to compare their reactivity effects.

We observed that they produced negative reactivity insertions not exactly linearly proportional to the angles of inclination of the assemblies when the flowering is around the central assembly. In fact, these deformations do not affect uniformly the geometry of the core because the expansions are more significant in the upper part of the core.

The linearity is restored in the case of flowering around a more external assembly (fifth ring assembly).

As we would expect, for the same definition of flowering levels, if the flowering source is around the central assembly, the effect on reactivity is higher than in the case of flowerings around more decentralized assembly. This is due to the different importance of the regions where the deformations occurs. As regards the ability of Cast3m in calculating the k_{eff} , at the end of the analysis, we can say that it depends strongly on the number of blocks used to define axially the assemblies.

Increasing the number of axial blocks describing the single assembly we can compute k_{eff} more and more converging to the one resulted from TRIPOLI.

Nevertheless, for our type of analysis, the discrepancy of some hundreds of pcm between the k_{eff} values computed with these two codes is acceptable (see Appendix).

CNTT, implemented in Cast3m in order to make criticality calculations on deformed geometries, is based on the mesh deformation method which consists in modifying the shape of the cells of the mesh in case of deformation and calculating their characteristic parameters as function of the new concentration of sodium in the cells, induced by the deformation.

Cast3m, despite that it is based on a simplified model and on diffusion theory, is capable of estimating the reactivity effects caused by the simpler deformations (compactions and straight widenings) with a good accuracy.

The percentage estimation errors resulted from Cast3m in these cases are reported in Table 5.1.

Table 5.1: Percentage estimation errors observed using Cast3m to calculate reactivity effects induced by simple types of deformation

Type of deformation	$\% {\it Estimation \ error} \ \Delta ho \ {\it Cast 3m15}$	$\% \ {\it Estimation \ error} \ \Delta ho \ {\it Cast3m23}$	
Compactions	10% - $14%$	3% - 7%	
Widenings	13% - 17%	6% - 10%	

The irregular deformations occurring during the post-explosion transient and high level flowering configurations are complex to simulate in Cast3m because of the difficulty introduced by these large irregularities in the calculations of the cross sections of deformed cells. For this reason, Cast3m is not reliable in computing the $\Delta \rho$ which undergo the system when its configuration is relevantly changed like in these cases.

In order to make Cast3m more reliable also for these types of deformations, it could be helpful to add cell models with higher concentration of sodium interassemblies inside, representing more points for performing the interpolations, needed to compute the cross sections.

The improvement in the performance of Cast3m could be very important in order to may rely on a code requiring low computational cost, in order to perform fastly and at the same time accurately mechanical-neutronic coupling.

The realization of the tests contained in this analysis was clearly dependent on the time needed by the simulations to reach convergence (especially regarding TRIPOLI calculations whose computational cost is quite expensive in terms of involved processors) compared to the six months available for the internship in CEA. For this reason, it was not possible to implement some features which could have led to improvements in the study such as the advantage of using more blocks to define the assemblies undergoing more complex deformations on TRIPOLI and also the possibility of testing the same deformed configurations of a core having compositions of the assemblies corresponding to an higher value of burnup. It would be worthwhile to compare the reactivity effects induced by the same deformations of the geometry on a core at two different burnup levels. Moreover, in order to complete the study, it would be interesting to apply to the tested configurations two other approaches available to perform this type of calculations :

- another mesh deformation method based on the use of irregular cells to take into account the deformation of the geometry, implemented in ERA-NOS by M. Gentili [26];
- mesh projection method developed in APOLLO3 (deterministic code) by C. Patricot during his PhD thesis [9] on the basis of C. Labarta's analysis of the effects of static flowerings of the Phenix core, realized during her internship in CEA Saclay [24].

As further hint for future works involving the study of the effects induced on the neutronics of a reactor core by deformations of its geometry, we could identify the benefits resulting from taking into consideration other interesting quantities such as the maximum and average value of the neutron flux but also the local (in some important locations) and global absorption and fission rates.

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List of Figures

1.1	The 4 generations of reactor $designs^{[4]}$	6
2.1	GFR design ^[11]	12
2.2	MSFR power plant scheme ^[13] \ldots \ldots	13
2.3	LFR reference designs: BREST-OD-300 (top), ELFR (left) and	
	SSTAR $(right)^{[11]}$	14
2.4	Thermal versus fast reactor neutron spectrum ^[14] $\ldots \ldots \ldots$	15
2.5	U-235 microscopic cross sections at 300 K	16
2.6	U-238 microscopic cross sections at 300 K	16
2.7	Pu-239 microscopic cross sections at 300 K	17
2.8	Number of neutrons released per fission ^[6] $\ldots \ldots \ldots \ldots \ldots$	19
2.9	Comparison between Pu-239 production cross section (blue line)	
	and Pu-239 fission cross section (red line) ^[6] $\ldots \ldots \ldots \ldots$	20
2.10	Transmutation probabilities expressed in $percentages^{[1]}$	22
2.11	Typical LWR and SFR lattice schemes	24
2.12	Example of typical SFR fuel assembly structure (FFTF fuel assembly	$^{[17]} 24$
2.13	SFRs configurations: pool type and loop type ^{$[16]$}	26
2.14	Core restraint system details ^[18] \ldots \ldots \ldots \ldots \ldots	30
3.1	Example of typical SFR core design ^[21] \ldots \ldots \ldots	33
3.2	Axial composition of model core assemblies	34
3.3	Simplified fuel pins composition scheme	35
3.4	TRIPOLI-4 [®] modeling scheme seen on the axial plane	36
3.5	Schematic representation of steps followed for TRIPOLI native	07
0.0	deformed geometry file	37
3.6	Inner core fuel assembly divided into mobile blocks	38
3.7	I RIPOLI-4° moving blocks modeling representation	39
3.8	View of the simulated geometry from above	40
3.9	View of the simulated geometry from the central axial plane	40
3.10 9.11	Detailed view of the external boundary from above	41
0.11 9.10	Cast2m mash of accompliant and inter accompliant axial plane	41
0.12	partial configuration Right: deformed configuration ^[9]	12
2 1 2	Avial view of the deformed assembly modeled with $Cast 2m^{[9]}$	40
2.10	Example of homogenized much deformation in Cast $3m^{[9]}$	40
3.14	Different cells homogeneization with different inter-assemblies sodium	-1-1 0
0.10	fractions	
3 16	Definition of deformed geometry in Cast3m	46
3.17	Visualization of core compaction	47
0.11	ristanzation of core compaction	- 1

3.18	Visualization of core widening	47
3.19	Visualization of core flowering from the center	49
3.20	Visualization of core flowering around an assembly belonging to	
	the fifth ring	50
4.1	Reactivity changes due to compaction of the core	54
4.2	Reactivity insertion due to compaction of the core	55
4.3	Reactivity changes due to widening of the core	57
4.4	Reactivity insertion due to widening of the core	58
4.5	Comparison of reactivity insertion due to same levels of com-	
	pactionwidening	58
4.6	Reactivity changes due to 6MJ explosion on the top of the core .	60
4.7	Reactivity insertion due to 6MJ explosion on the top of the core	61
4.8	Reactivity changes for core flowering starting from the center	63
4.9	Reactivity insertion for core flowering starting from the center	64
4.10	Reactivity changes for core flowerings around the central assem-	
	bly and around a fifth ring assembly	65
4.11	Reactivity insertion for core flowering with the source on a fifth	
	ring assembly	66

List of Tables

2.1	Capture-to-fission ratio of U-235 and Pu-239 at different energies according to JENDL 4.0	18
$2.2 \\ 2.3$	Advantageous features of the two SFRs configurations Sodium-Cooled Fast Reactors World Wide Experience	$25 \\ 27$
$3.1 \\ 3.2 \\ 3.3$	Definition of the levels of compaction	47 48
$3.4 \\ 3.5$	our static criticality calculations	48 49 50
4.1	Convergence study to choose the number of points of Cast3m	00
4.0	axial mesh	53
4.2	figurations computed with TRIPOLI reference code	54
4.3	Comparison between reactivity effects computed with TRIPOLI and Cast3m (compactions)	56
4.4	Reactivity coefficient and reactivity effects of the widening con- figurations computed with TRIPOLI reference code	56
4.5	Comparison between reactivity effects computed with TRIPOLI	50
4.6	Reactivity coefficient and reactivity effects due to deformations of the transient following a 6MJ explosion on top of the core,	97
4.7	computed with TRIPOLI reference code	59 60
4.8	Reactivity effects due to increasing flowerings around the central	00
4.9	assembly	62 64
4.10	Reactivity changes computed with TRIPOLI in case of flowerings	65
4.11	Comparison between reactivity effects computed with TRIPOLI and Cast3m (flowering around a fifth ring assembly)	66

5.1	Percentage estimation errors observed using Cast3m to calculate reactivity effects induced by simple types of deformation	69
A.1	Multiplication factor, reactivity and reactivity effect values computed with TRIPOLI-4 [®] due to compactions $\dots \dots \dots \dots$	77
A.2	Multiplication factor, reactivity and reactivity effect values com- puted with Cast3m (15-point axial mesh) due to compactions	77
A.3	Multiplication factor, reactivity and reactivity effect values com- puted with Cast3m (23-point axial mesh) due to compactions	78
A.4	Multiplication factor, reactivity and reactivity effect values com- puted with TRIPOLI-4 [®] due to widenings	78
A.5	Multiplication factor, reactivity and reactivity effect values com- puted with Cast3m (15-point axial mesh)due to widenings	78
A.6	Multiplication factor, reactivity and reactivity effect values com- puted with Cast3m (23-point axial mesh) due to widenings	79
A.7	Multiplication factor, reactivity and reactivity effect values com- puted with TBIPOLI-4 [®] due to 6MJ explosion transient defor-	
Δ 8	mations	79
11.0	puted with Cast3m (15-point axial mesh) due to 6MJ explosion transient deformations	80
A.9	Multiplication factor, reactivity and reactivity effect values com-	80
A 10	transient deformations	80
A.10	puted with TRIPOLI-4 [®] due to flowerings around central assembly	81
A.11	puted with Cast3m (15-point axial mesh) due to flowerings around	01
A.12	Multiplication factor, reactivity and reactivity effect values com-	81
	puted with Cast3m (23-point axial mesh) due to flowerings around central assembly	81
A.13	Multiplication factor, reactivity and reactivity effect values com- puted with TRIPOLI-4 [®] due to flowerings around a fifth ring	
A.14	Assembly	82
	puted with Cast3m (15-point axial mesh) due to flowerings around a fifth ring assembly	82
A.15	Multiplication factor, reactivity and reactivity effect values com- puted with Cast3m (23-point axial mesh) due to flowerings around	
	a fifth ring assembly	82

Appendix A

Tables of reactivity and k_{eff} values computed with TRIPOLI-4[®] and CAST3M

A.1 Compactions

A.1.1 TRIPOLI-4[®] results

Table A.1: Multiplication factor, reactivity and reactivity effect values computed with TRIPOLI-4 $^{\circledast}$ due to compactions

	k_{eff}	σ [pcm]	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.06449	4	6058	-
Comp2	1.06510	5	6112	54
Comp4	1.06636	5	6223	165
Comp8	1.06898	5	6452	394
Comp14	1.07292	5	6796	738

A.1.2 Results of CAST3M using 15 and 23-point axial mesh

Table A.2: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (15-point axial mesh) due to compactions

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0316	3066	-
Comp2	1.0321	3115	49
Comp4	1.0332	3212	146
Comp8	1.0353	3407	340
Comp14	1.0384	3700	634

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0590	5569	-
Comp2	1.0596	5621	53
Comp4	1.0607	5727	158
Comp8	1.0631	5939	370
Comp14	1.0668	6258	690

Table A.3: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (23-point axial mesh) due to compactions

A.2 Widenings

A.2.1 TRIPOLI-4[®] results

Table A.4: Multiplication factor, reactivity and reactivity effect values computed with TRIPOLI-4[®] due to widenings

	k_{eff}	σ [pcm]	ho [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.06449	4	6058	-
Gerb2	1.06381	6	5998	-60
Gerb4	1.06252	6	5884	-174
Gerb8	1.05994	5	5655	-403
Gerb14	1.05620	5	5321	-737

A.2.2 Results of CAST3M using 15 and 23-point axial mesh

Table A.5: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (15-point axial mesh)due to widenings

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0316	3066	-
Gerb2	1.0311	3016	-50
Gerb4	1.0301	2918	-148
Gerb8	1.0280	2721	-346
Gerb14	1.0249	2426	-641

	keff	rho [pcm]	deltarho [pcm]
Nominal	1.0590	5569	-
Gerb2	1.0584	5515	-54
Gerb4	1.0572	5408	-161
Gerb8	1.0548	5195	-374
Gerb14	1.0513	4876	-693

Table A.6: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (23-point axial mesh) due to widenings

A.3 Deformations recorded during transient following 6MJ explosion on top of the core

A.3.1 TRIPOLI-4[®] results

Table A.7: Multiplication factor, reactivity and reactivity effect values computed with TRIPOLI-4[®] due to 6MJ explosion transient deformations

	k_{eff}	σ [pcm]	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.06449	4	6058	-
6MJ_1	1.05655	5	5353	-706
6MJ_2	1.05497	5	5211	-847
6MJ_3	1.06077	5	5729	-330
6MJ_4	1.06619	5	6208	150
6MJ_maxcomp	1.06683	5	6265	206
6MJ_5	1.06682	5	6263	205
6MJ_6	1.06334	6	5957	-102
6MJ_7	1.06089	6	5739	-319
6MJ_8	1.06040	5	5696	-363
6MJ_9	1.06176	5	5816	-242
6MJ_10	1.06337	5	5959	-99

A.3.2 Results of CAST3M using 15 and 23-point axial mesh

Table A.8: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (15-point axial mesh) due to 6MJ explosion transient deformations

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0316	3066	-
6MJ_1	1.0279	2717	-349
6MJ_2	1.0267	2604	-462
6MJ_3	1.0303	2939	-128
6MJ_4	1.0332	3216	149
6MJ_maxcomp	1.0334	3236	170
6MJ_5	1.0336	3253	187
6MJ_6	1.0314	3041	-26
6MJ_7	1.0300	2913	-154
6MJ_8	1.0298	2891	-176
6MJ_9	1.0305	2959	-107
6MJ_10	1.0314	3047	-20

Table A.9: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (23-point axial mesh) due to 6MJ explosion transient deformations

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0590	5569	/
6MJ_1	1.0545	5169	-400
6MJ_2	1.0532	5052	-517
6MJ_3	1.0571	5402	-166
6MJ_4	1.0605	5708	139
6MJ_maxcomp	1.0611	5760	192
6MJ_5	1.0611	5760	191
6MJ_6	1.0583	5506	-63
6MJ_7	1.0567	5370	-199
6MJ_8	1.0565	5347	-222
6MJ_9	1.0573	5423	-146
6MJ_10	1.0584	5522	-47

A.4 Flowerings with source at the center of the core

A.4.1 TRIPOLI-4[®] results

Table A.10: Multiplication factor, reactivity and reactivity effect values computed with TRIPOLI-4[®] due to flowerings around central assembly

	k_{eff}	σ [pcm]	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.06449	4	6058	-
Flower_0.5	1.06316	7	5941	-118
Flower_1	1.06200	5	5838	-220
Flower_1.5	1.06082	13	5733	-325
Flower_3	1.05753	6	5440	-618
Flower_5	1.05358	14	5086	-973

A.4.2 Results of CAST3M using 15 and 23-point axial mesh

Table A.11: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (15-point axial mesh) due to flowerings around central assembly

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0316	3066	-
Flower_0.5	1.0307	2983	-83
Flower_1	1.0300	2911	-155
Flower_1.5	1.0296	2876	-190
Flower_3	1.0292	2835	-231
Flower_5	1.0290	2822	-244

Table A.12: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (23-point axial mesh) due to flowerings around central assembly

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0590	5569	-
Flower_0.5	1.0578	5466	-103
Flower_1	1.0569	5381	-188
Flower_1.5	1.0564	5343	-226
Flower_3	1.0560	5300	-269
Flower_5	1.0559	5292	-277

A.5 Flowerings with source located at the position of a fifth ring assembly

A.5.1 TRIPOLI-4[®] results

Table A.13: Multiplication factor, reactivity and reactivity effect values computed with TRIPOLI-4[®] due to flowerings around a fifth ring assembly

	k_{eff}	σ [pcm]	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.06449	4	6058	-
Flower_1_66	1.06269	13	5899	-159
Flower_3_66	1.05906	13	5577	-481
Flower_5_66	1.05550	13	5258	-800

A.5.2 Results of CAST3M using 15 and 23-point axial mesh

Table A.14: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (15-point axial mesh) due to flowerings around a fifth ring assembly

	k_{eff}	ρ [pcm]	$\Delta \rho \ [pcm]$
Nominal	1.0316	3066	-
Flower_1_66	1.0301	2922	-144
Flower_3_66	1.0294	2856	-210
Flower_5_66	1.0294	2852	-214

Table A.15: Multiplication factor, reactivity and reactivity effect values computed with Cast3m (23-point axial mesh) due to flowerings around a fifth ring assembly

	k_{eff}	ho (pcm)	$\Delta \rho$ (pcm)
Nominal	1.0590	5569	-
$Flower_1_66$	1.0571	5399	-170
Flower_3_66	1.0563	5327	-242
Flower_5_66	1.0562	5324	-245