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## To cite this version:

G. Rimpault, P. Dufay, J. Tommasi, F. Mellier. looking forward for a masurca experimental programme genesis in support to the astrid sfr core. PHYSOR 2018 Reactors Physics paving the way towards more efficient systems, Apr 2018, Cancun, Mexico. hal-02416226

**HAL Id: hal-02416226**

**https://hal.science/hal-02416226v1**

Submitted on 17 Dec 2019

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*PHYSOR 2018: Reactors Physics paving the way towards more efficient systems*
Cancun, Mexico, April 22-26, 2018

# LOOKING FORWARD FOR A MASURCA EXPERIMENTAL PROGRAMME GENESIS IN SUPPORT TO THE ASTRID SFR CORE

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**ABSTRACT**

The characteristics of the ASTRID core (Advanced Sodium Technological Reactor for Industrial Demonstration) must be determined according to well-defined calculation conditions and accuracies.
In order to achieve that goal, several steps are required such as mastering the computation procedures, checking the nuclear data in use and bringing evidence with experiments of the validity of the produced values and their uncertainties.
Past experiments will allow a quantification of the uncertainties for core characteristics such as βeff or Doppler, a reduction of the uncertainties due to nuclear data (through assimilation) which might remain insufficient. Therefore, past experiments will not cover all requirements either because of the level of uncertainties or because it will not cover ASTRID core specificities such as sodium void reactivity, control rod worth and power map distribution over the cycle.
The MASURCA facility is currently under refurbishment until 2023 and will exhibit improved performances. The future GENESIS experimental programme of the MASURCA facility will take advantage of advanced measurement techniques, on one hand, and the possibility of using fuel with degraded Pu vector to develop a solid knowledge of ASTRID core characteristics.

KEYWORDS: SFR, ASTRID, MASURCA

## 1. INTRODUCTION

The characteristics of the ASTRID core[1] (Advanced Sodium Technological Reactor for Industrial Demonstration) must be determined according to well-defined calculation conditions and accuracies. To this end, Verification / Validation / Uncertainty Quantification (VVUQ) of the Neutron Calculation Tools are required.
This plan consists in several steps, with particular emphasis on the particularities of the reference CFV[2] core, covering the following items:

- recommended APOLLO3®[3] calculation procedures, their checks and validations,
- recommended nuclear data[4] associated with matrices of covariances[5] and
- proof of the achieved best estimate values plus uncertainties[6].

This last step uses the integral experiments to reinforce the uncertainties of the calculation scheme and the nuclear data and to quantify the predictive value:

- Quantification of the uncertainties of the characteristics of the ASTRID CFV core on dedicated experiments such as βeff[7] or Doppler[8].
- Analysis of experiments in critical BFS installations (Phases I and III in the BFS-2 assembly and Phases II and IV in the BFS-1 assembly)[9] and MASURCA (GENESIS program) in support of the specific characteristics of the cores ASTRID, including the voiding effect, the hot-spot and control rod reactivities (S-like curve),
- Analyses of sample reactivity worths and irradiated samples to master core characteristics changes due to depletion.

This paper describe the actions that are required to quantify uncertainties for core characteristics such as βeff or Doppler, a reduction of the uncertainties due to nuclear data (through assimilation) and a validation of core heterogeneities such as inner axial blanket or sodium plenum.

Beyond 2023, the quantification of the uncertainties of the neutron tools for ASTRID safety dossier will involve expanding the experimental base with the future programme GENESIS of the MASURCA facility, currently under refurbishment. This programme is focused on ASTRID core specificities such as sodium void reactivity, control rod worth, power map distribution with but also on nuclear data generic requirements for fast reactors.

The MASURCA facility will exhibit, after refurbishment, improved performances. The future GENESIS experimental programme of the MASURCA facility will take advantage of advanced measurement techniques with in particular reduced background noise enabling better neutron noise measurements and sample worth measurement using the open loop technique.

The reduction of the uncertainties of the core characteristics of the core will hence be possible for such important core characteristics such as βeff or burn up swing. The possibility of using fuel with degraded Pu vector will address a better knowledge of ASTRID core characteristics which are out of past experience either in critical facilities or reactors.

The paper presents in chapter 2, the ASTRID CFV core which requires a safety dossier with solid demonstration ever done before. The paper presents in chapter 3, the MASURCA facility with its improved features, the existing fuel stockpile and the details of the measurement techniques foreseen. In Chapter 4, the various experiments conducted in the past in the MASURCA facility are presented as they are having a dedicated role on some specific parameters and for reducing nuclear data uncertainties. Chapter 5 presents GENESIS plans in MASURCA for covering such diverse characteristics of the core such as voiding effect, hot-spots and control rod reactivities (S-like curve) with particular attention to their behavior with changes in the Pu vector.

## 2. THE CFV CORE OF ASTRID

The ASTRID CFV core ("Low Sodium Void core ") contains innovations to achieve a zero sodium void reactivity effect at end-of-cycle. A fertile plate in the inner core increases the axial flux gradient at the boundary of the core and the sodium plenum. The sodium plenum acts as a neutron reflector in normal operation and increases the axial leakage when voided. The absorbing material on the top of the plenum region prevents in this case the neutrons to come back to the core. The external core is higher than the inner core in order to increase the neutron leakage and limit the core radius increase.

Design changes were made on the ASTRID CFV-V3 core, giving rise to a new version of the core, the ASTRID CFV-BD core, which differs substantially from the previous cores, in particular by the isotopic vector of the Pu in the fuel and hence the content of Pu (Table 1). The MOX fuel now comes from a reprocessed PWR MOX spent fuel while it was previously coming from a PWR UOX spent fuel.

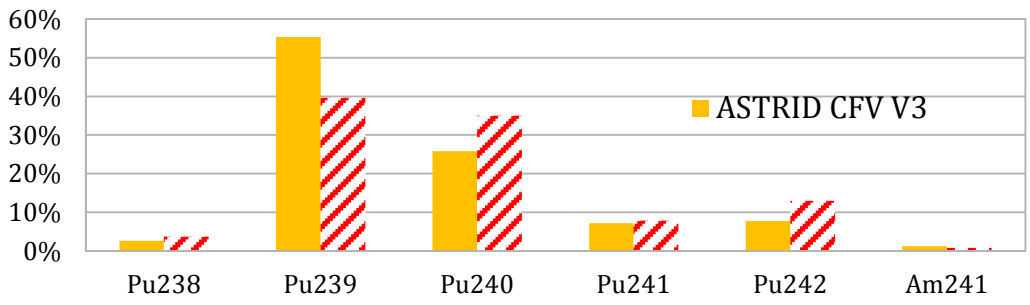


**Figure 1: Comparison of Pu vectors and Pu contents for the CFV-V3 and CFV- BD cores**

The separation of the leakage and the non-leakage components of the reactivity worth in transport theory[10] is performed with the following equation (1):

$$LC = \sum_g \delta\Sigma_t^g \left( \frac{1}{(4\pi^2)} \int_{4\pi} \Psi_g d\vec{\Omega} \int_{4\pi} \Psi_g^+ d\vec{\Omega} - \frac{1}{4\pi} \int_{4\pi} \Psi_g \Psi_g^+ d\vec{\Omega} \right) + \text{P1 scattering terms} \qquad (1)$$

where $\Psi_g$ and $\Psi_g^+$ are respectively the direct and adjoint angular fluxes.

The highly degraded Pu fuel (35% Pu240 content) makes it necessary to increase the Pu content by around 3% to compensate for the loss of reactivity and this tends also to degrade the sodium void reactivity effect as seen in the following Table 1, column Total.

**Table 1: Comparison of the void reactivity effect for the CFV-V3 and CFV-BD cores**

| (in pcm) | Total | CC | LC |
|---|---|---|---|
| CFV-V3 | -920 | 2362 | -3282 |
| CFV-BD | -529 | 2760 | -3290 |

The overall leakage component (LC) in Table 1 is not modified (-0.2%) when moving from one core to the other. However, the non-leakage component or central component (CC) is more important (+14.4%) in this CFV-BD core than in the CFV-V3 core. This is due to steeper slope of adjoint function due to a change of the ratio of fission over capture induced by the Pu240 threshold fission cross section. This is quantified by $\zeta$ defined as the ratio of the average importance of fission neutrons and leaking neutrons. It therefore characterizes the more or less steep slope of the neutron importance between the corresponding average energies

The optimisation of the core geometry has not been done in the same way for the CFV-BD than the CFV-V3, which leads to some changes in the leakage distribution. However, the overall leakage component is not modified (-0.2%) when moving from one core to the other. This is important to notice as it has been demonstrated through a recent PhD work that there is a link between the power shape and leakage distribution. This will further investigated in the Paul Dufay PhD work.

The Pu vector deteriorates the core reactivity feedbacks and in particular on the SVRE. This negative impact is being handled without difficulties by the CFV-BD core. However, this Pu vector change induces a significant uncertainty change. The uncertainties have now been calculated using the sandwich formula at the beginning of the cycle at equilibrium and for nominal operating conditions using the JEFF3.2 library and the COMAC-V2 covariance matrix. Results are presented in table 2 using the SPT for the critical mass and in table 3 using the generalized perturbation equivalent method (EGPT) for the sodium void reactivity effect (SVRE).

**Table 2 : Uncertainties (in pcm) on the CFV-BD $k_{eff}$ using COMAC-V2 (*low contributions have been skipped*)**

| Isotope | Fission | Capture | Elastic | Inelastic | NxN | Nu | Fission Spectrum | Total |
|---|---|---|---|---|---|---|---|---|
| O16 | 0 | 110 | 50 | 0 | 0 | 0 | 0 | 120 |
| Na23 | 0 | 20 | 10 | 10 | 0 | 0 | 0 | 30 |
| Fe56 | 0 | 80 | 30 | 50 | 0 | 0 | 0 | 100 |
| Ni58 | 0 | 50 | 0 | 0 | 0 | 0 | 0 | 50 |
| U238 | 100 | 340 | 40 | 379 | 30 | 70 | 60 | 529 |
| Pu238 | 10 | 20 | 0 | 0 | 0 | 80 | 40 | 100 |
| Pu239 | 300 | 150 | 10 | 50 | 10 | 100 | 210 | 409 |
| Pu240 | 669 | 300 | 0 | 60 | 10 | 20 | 170 | 619 |
| Pu241 | 70 | 90 | 0 | 10 | 0 | 70 | 150 | 200 |
| Pu242 | 40 | 50 | 0 | 20 | 0 | 20 | 40 | 80 |
| TOTAL | 739 | 290 | 40 | 379 | 30 | 170 | 320 | 959 |

**Table 3 : Uncertainties (in cents) on the CFV-BD SVRE using COMAC-V2 (*low contributions have been skipped*) ((βeff=371pcm, 1 cents = 1/100 $)**

| Isotope | Fission | Capture | Elastic | Inelastic | NxN | Nu | Fission Spectrum | Total |
|---|---|---|---|---|---|---|---|---|
| O16 | 0.00 | 1.01 | 0.38 | 0.03 | 0.00 | 0.00 | 0.00 | 1.08 |
| Na23 | 0.00 | 3.10 | 3.05 | 2.29 | 0.01 | 0.00 | 0.00 | 4.92 |
| Fe56 | 0.00 | 2.40 | 1.14 | 0.41 | 0.00 | 0.00 | 0.00 | 2.68 |
| Ni58 | 0.00 | 0.50 | 0.10 | 0.03 | 0.00 | 0.00 | 0.00 | 0.51 |
| U238 | 1.80 | 4.71 | 0.34 | 3.18 | 0.29 | 0.72 | 0.74 | 6.06 |
| Pu238 | 0.13 | 0.64 | 0.02 | 0.02 | 0.00 | 0.33 | 0.16 | 0.76 |
| Pu239 | 8.94 | 3.47 | 0.10 | 0.10 | 0.01 | 1.37 | 0.67 | 9.71 |
| Pu240 | 5.50 | 3.63 | 0.15 | 0.55 | 0.01 | 0.19 | 1.52 | 6.74 |
| Pu241 | 1.72 | 1.14 | 0.03 | 0.06 | 0.01 | 0.66 | 0.76 | 2.29 |
| Pu242 | 0.46 | 1.80 | 0.06 | 0.08 | 0.00 | 0.23 | 0.10 | 1.87 |
| TOTAL | 10.80 | 8.32 | 3.31 | 3.90 | 0.29 | 1.74 | 1.98 | 14.80 |

The main conclusion of that comparison is the significant increase of the Pu240 contribution while Pu239 and U238 contributions remain high. Particular attention will be given to this Pu vector content in the GENESIS experimental programme but also the impact it has on the axial flux distribution since it affects significantly the axial leakage component.

## 3. The MASURCA Facility

The MASURCA[1] reactor, operated by the French Alternative Energies and Atomic Energy Commission (CEA) at Cadarache, is a zero-power critical research facility for fast reactor applications. The MASURCA facility (figure 2) is dedicated to the neutronic studies of fast lattices using plutonium fuels. It is an air cooled reactor operating at a maximum power of 5 kW and a flux level up to $2x10^{10}$ n/cm$^2$/s.

[1] MAquette SURgénérateur de CAdarache

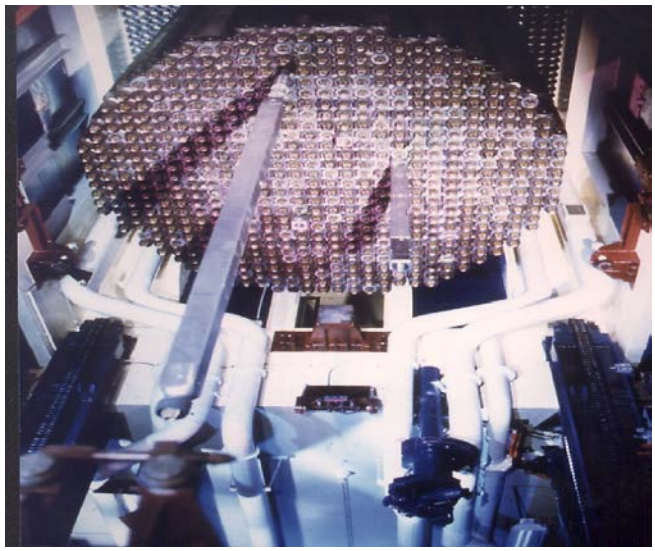

**Figure 2: View of MASURCA facility**

The materials of the core are contained in cylinder or squared rodlets. These rodlets (Figure 3a) are put into wrapper tubes having a square section (4 x 4 inches) and about 3 meters in height. The use of platelets (Figure 3b) from the ZEBRA and SNEAK facilities is also possible in which case the wrapper tube has a square section (2 x 2 inches) and about 3 meters in height.

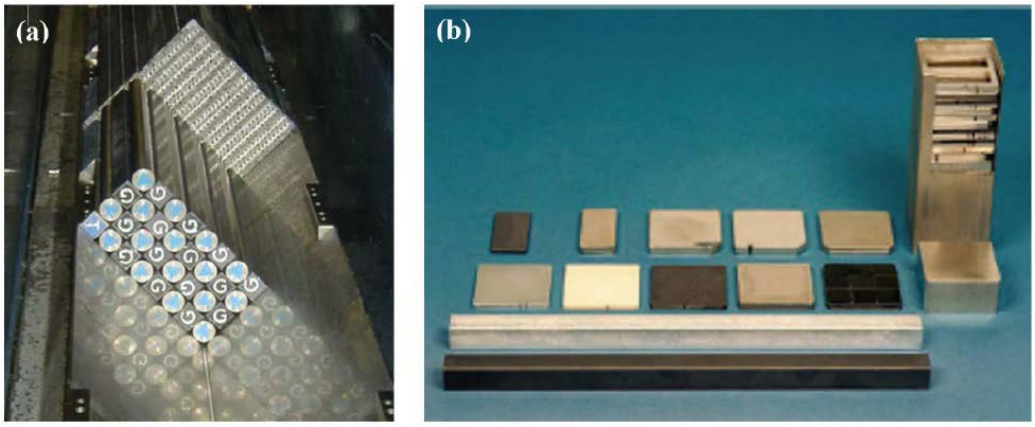


Figure 3: MASURCA reactor available stockpile. (a) Pins. (b) Plates.

The reactivity is controled by absorber rods whatever the core types and sizes. These safety rods (SR) are loaded with fuel materials in their lower part, so that the core is homogeneous when the rods are withdrawn.

MASURCA had its first criticality in 1966. Experimental programmes were devoted to the studies conducted in support of the PHENIX, SUPERPHENIX and EFR (European Fast Reactor) projects. The first experimental programmes (1966-75) were designed so as get a better knowledge of nuclear data which was and remains of poor quality. Different cores with different Pu contents were loaded so as to vary energy spectrum and hence sensitivities to nuclide cross sections.

The BERENICE (Beta Effective Reactor Experiment for a New International Collaborative Evaluation) programme was devoted to the measure of the effective fraction of delayed neutrons (βeff) by various methods: Cf source and the noise techniques methods. It included two cores: one with U enriched fuel (basic R2 cell), the other using mixed fuel (basic ZONA2 cell). This program was part of a more general

collaboration for the measurement of the βeff in fast reactors with different institutions in addition to the France, from United States, Belgium, Russia and Japan.
The CIRANO (1994-97) went in support of Pu burner cores as they were designed in the CAPRA programme (no blanket, higher enrichment, larger dilution).
The COSMO program (1998-1999) aimed at studying the Physics of the irradiation of LLFP targets in moderated sub-assembly at the periphery of fast reactors. The effects of various moderators (11B4C,CaH, ZrH2), have been investigated. This program helped in particular to the preparation of the ECRIX experiments that have been loaded in the PHENIX reactor in 2004.
From 1999 to 2005, the MUSE program allowed to study configurations with reactivity level representative of those envisaged for future ADS.

## 4. The ASSETS of PAST EXPERIMENTS

### 4.1. Nuclear Data Assimilation

Integral experiments have been analyzed recently with the TRIPOLI4[11] code so as to produce C/E values without modeling approximation biases. The analysis of these experiments (C/E and analyzes of sensitivities to nuclear data) allows a feedback to the nuclear data libraries and their associated covariances.

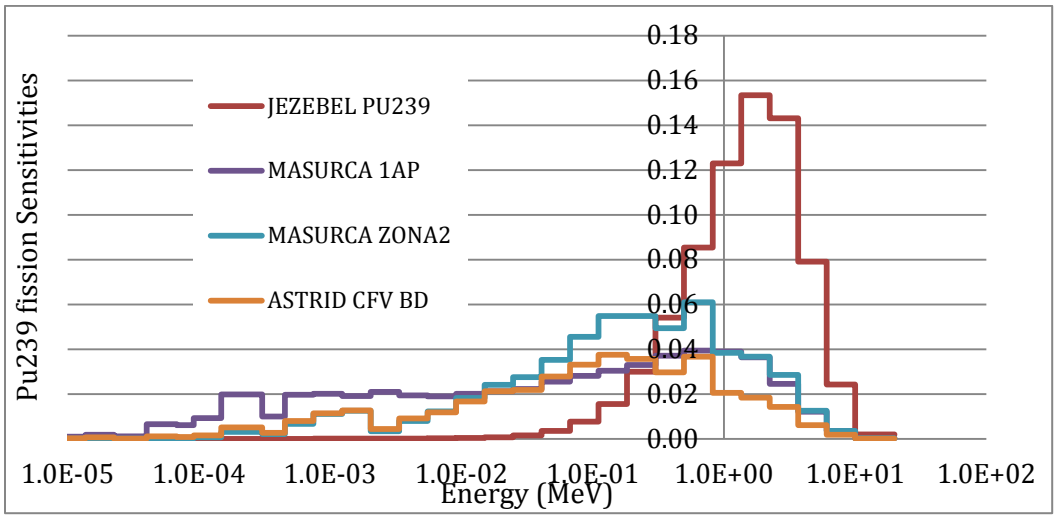


Figure 4 Critical mass sensitivity profiles of the Pu239 fission cross-section of for JEZEBEL Pu239, MASURCA 1AP, MASURCA ZONA2 and ASTRID CFV

The MASURCA 1AP core, the PRE-RACINE cores[12] using fuels with different Pu vectors, the ZONA2 core of the BERENICE[13] and CIRANO[14,15] programmes and also the JEZEBEL core of the ICSBEP data base offer a complementary information on Pu239 fission cross sections as it is illustrated with sensitivity profiles in Figure 4. More core analyses are necessary to provide complementary information and eventually redundant ones in order to reduce some components of neutron balance for the ASTRID CFV-BD core either at Beginning of Cycle (BoC) or at End of Cycle (EoC). Sample reactivity worths measured with oscillation techniques and analyses of irradiated sample will be used in another PhD work with similar objectives. Nevertheless, there is no specific experiments devoted to ASTRID CFV-BD in this data base and the reduction of uncertainties will be limited and will have to be complemented by new measurements.

### 4.2. βeff characterization

The BERENICE experimental programme of MASURCA devoted to the βeff characterization has been revisited[7] using the most recent Monte Carlo developments i.e. the IFP method in TRIPOLI4[16].

Experimental corrections on βeff measurements were hence possible with a detailed core modeling. Also GPT method was used to calculate the impact of nuclear data uncertainty on this parameter. The C/E is very low with JEFF3.2 (1.2% +/- 3.6%) for the BERENICE-ZONA2 noise measurement of the ZONA2 core of the BERENICE programme loaded with MOX fuel. The calculation of uncertainties due to nuclear data on this kinetic parameter gives a value of 2.8% with COMAC-V2 which is lower than the experimental uncertainty. New measures with reduced uncertainties are required for a reduction of the predictive uncertainty. These are planned during the GENESIS programme.
The predictive calculation on βeff is used for analysing Sodium void reactivity effect (SVRE) measurements which are performed using the dollar unit (the reactivity variation normalized by the effective delayed neutron fraction).

### 4.3. Sodium Void Reactivity Effect characterization

The past SVRE experiments are analysed by splitting the sodium voiding reactivity effect between a central component and a leakage component. Special attentions are being paid to the MASURCA CIRANO and PRE-RACINE programmes, as they include different configurations of sodium voiding to exacerbate either one or the other of the two components. None of the SVRE past experiments are truly representative of central component of the ASTRID CFV core as this is illustrated on figure 3. Experiments such as PRE-RACINE programme have characteristics quite close to the ASTRID CFV-V3 core, but this is no longer the case for ASTRID CFV-BD version.

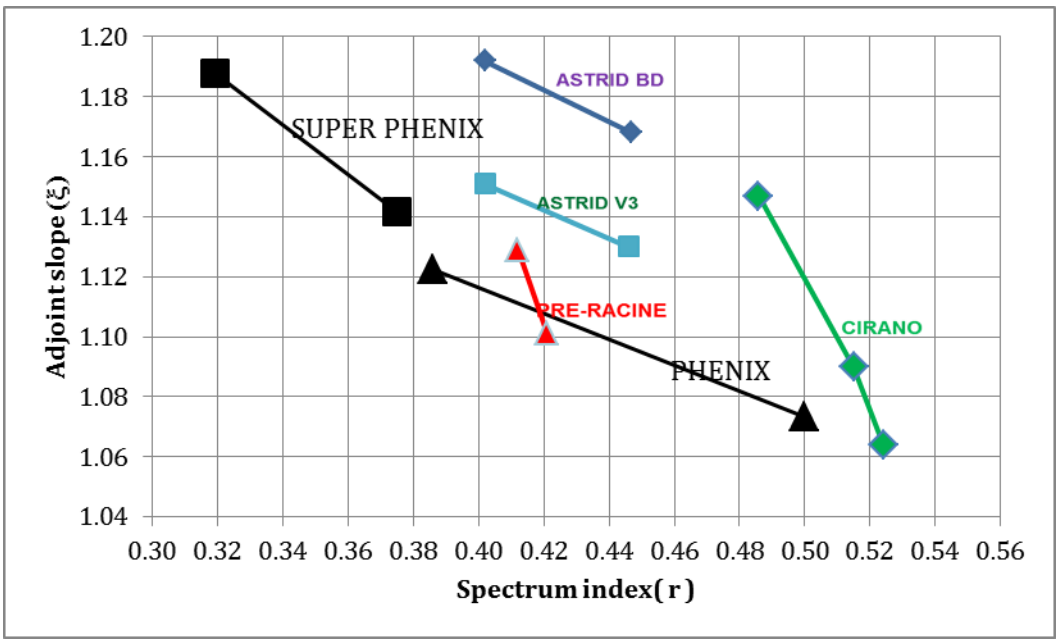


**Figure 5: Characteristics of the past Experimental and Reactor Cores**

The BFS experimental programme conducted at IPPE in collaboration with CEA have features very much linked to the ASTRID CFV core with experimental reactivities for nominal case but also various voided core and plenum regions and a control rod in the central position. However, the fuel is not at all representative with its high Pu239 content and several geometrical aspects are lacking such as an axial boron shield.
Hence, existing experiments will allow a quantification of the uncertainties for core characteristics such as βeff or Doppler, a reduction of the uncertainties due to nuclear data (through assimilation) but will not cover all requirements. It will not cover for instance ASTRID core specificities such as sodium void reactivity, control rod worth, power map distribution over the cycle as it will be demonstrated in the next chapter.

## 5. FUTURE MASURCA PLANS

### 5.1. Overall GENESIS experimental programme

ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) innovative core contains many axial and radial heterogeneities (in order to obtain a negative void coefficient) and interfaces that are challenging for current deterministic codes to simulate correctly.

A future programme called GENESIS[2] will be performed in support to the prototype ASTRID to validate the CFV core specificities such as sodium void reactivity, control rod worth, power map distribution. This multiyear programme scheduled from 2023 on, after the refurbishment of the facility, will give an important database on the ASTRID demonstrator specificities and nuclear data generic requirements for fast reactors.

For the CFV concept, because of the synergy in reducing the SVRE between the internal fertile plate displaced from the mid plane of the core, the sodium plenum and the absorbing zone above the plenum Finally, from the sensitivity of the SVRE to the composition of the core / plenum interface, new integral SVRE measurements under these conditions are necessary, separating and then combining the various effects. Indeed, there are only heterogeneous axial configurations with plate centered and without plenum, or configurations with plenum but without plate. Experimental configurations of the following specificities of the control rods in the CFV are also lacking:

- interaction between the control rod and the fertile plate in the inner core;
- Absorbent part of the control rods when located in the plenum, with strong variation of the neutron screening from the core between nominal condition and voided one;
- interaction of the control rod in case of asymmetric insertions and the presence of the fertile plate.

More generally, a measurement campaign representative of the efficiency of absorbents, including interaction effects in nominal conditions and through inadvertent removal event, is desirable to reliably quantify and reduce uncertainties to be attributed to hot spots in operational or accidental situations of the core.

The new integral experiments made necessary for the validation of tools on the specificities of the CFV core will be advantageously used to make precise and detailed measurements at the core / reflector or core / plenum interfaces, as well as in the reflector and the plenum and this either fully Na loaded or voided.

Finally, the ASTRID configuration includes an internal storage area and the presence of intermediate exchangers relatively close to the core. Effective neutron shielding - composed of MgO and B4C - minimizes the activation of secondary sodium. For a precise characterization of these neutron protections, specific experiments will be necessary.

Independently from these aspects, the use of most recent nuclear data evaluations has shown progresses which are consolidated and/or improved by the use of integral experiments assimilation techniques especially on capture/fission/inelastic $^{238}$U cross section, capture/fission/prompt neutron spectrum $^{239}$Pu data and all other heavy nuclei such as $^{240}$,$^{242}$Pu, $^{241}$Am. Nuclear data of those nuclides as well as those of Minor Actinides will be improved by using reactivity worth measurements which could be directly performed in the fast reactor MASURCA using an open loop oscillator technique. The open loop techniques[17] (pile oscillator method analyzed with harmonic analysis and inverse kinetics) could allow measurements of small worth sample reactivities of the order of 0.04 cents with standard deviations on the order of 0.003 cents.

Also absolute fission ratio measurements have the possibility of getting useful information as it has illustrated by the FCA-IX campaign[18]. These absolute fission chambers are rather large but detailed stochastic calculations are possible and give access to rather precise fission rate ratios. These will be tested during the AOSTA program of the TAPIRO installation (ENEA).

[2] Génération d'Expériences Neutroniques pour les Systèmes Innovants au Sodium

The miniature fission chamber measurements require a fast converter column calibration which is not independent of the fission sections to be measured. However, they have a relative interest when looking at the spectrum distribution within the core. Given the axial heterogeneity of the core, they will be valuable for understanding the physics of shape of the flux either in nominal conditions or in voided conditions.
The measurement of the kinetic characteristics of the GENESIS core is accurate only with the neutron noise method. This method requires that the neutron return by the reflective media is minimized and therefore requires a homogeneous core configuration with fertile covers (to be designed with TRIPOLI4). Greater accuracy of possible measurements associated with a renovation of MASURCA minimizing background noise.
Since ASTRID CFV unprotected events are less energetic than previous homogeneous cores, criticality issues in the course of these events need to be addressed more precisely. It is why a series of distorted configurations will be set up in a similar manner than previous SNEAK 12A & 12B experimental programmes[19] but for configurations more directly linked to these scenarios.
In order to limit the size of the reactor vessel, shielding is designed so as to be quite efficient to limit the damage dose to the reactor vessel and to the activation of the intermediate sodium through the heat exchanger. This requires the use of successive layers of MgO, SS and B4C and this with or without Internal Storage (SI). The SI storage is designed so as to limit the fuel transfer duration between the core and the fuel storage tank. Again there will be a significant part of the GENESIS programme devoted to this aspect.

### 5.2. GENESIS core characteristics and related SVRE

The SVRE measured in the MASURCA critical facility will take the usual form of successive substitutions of different sizes, in order to vary the relative importance of spectral and leakage components. They should also allow qualifying the sensitivity of the SVRE to variations in the Uranium plate (thickness, position) or the Na plenum (position, composition changes near the core / plenum interface). Therefore, two MASURCA configurations have been set up in order to address the various concerns associated to the sodium void reactivity effect (SVRE) of the ASTRID CFV BD core.
As mentioned in chapter 2, the SVRE can be split into two components:

- a central component which represents the positive contribution of the core regions and is due to the reduction in the slowing down process moving neutrons to higher energies where they have larger importance,
- a leakage component negative which comes from the change in mean free path of the neutrons. This negative component of the sodium void reactivity effect (SVRE) can be split into two intricate contributions: one due to core regions and the other ones due to subcritical media such as plenum and associated axial core heterogeneities.

Analyzing the dependencies on nuclear data and the calculation methods of each of the components makes it possible to appropriately understand how the experimental validation of the sodium void reactivity effect of the ASTRID CFV core can be done. The first one (GENESIS-0) uses fuel platelets to represent accurately the axial geometry of the ASTRID CFV BD core (Figure 6).

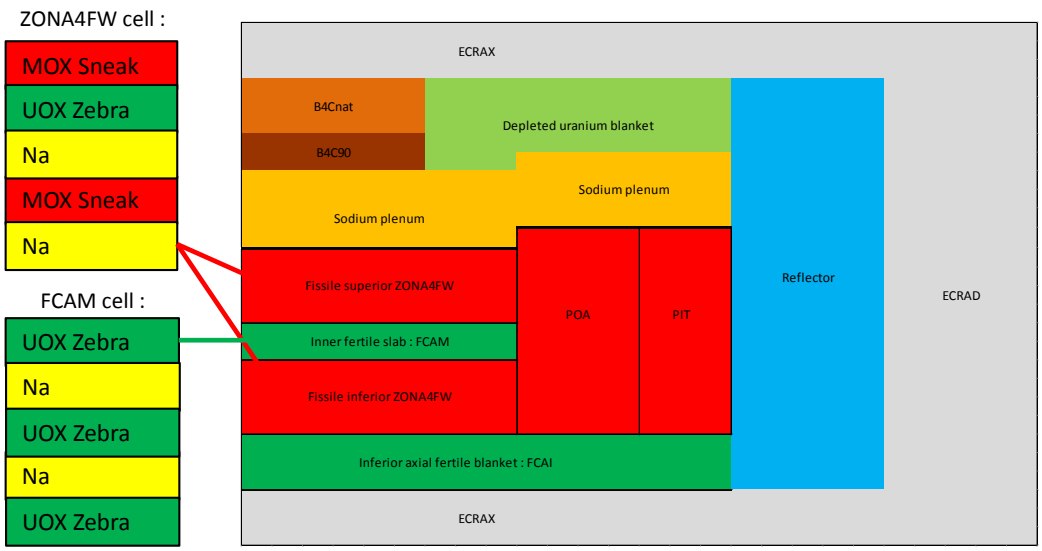


**Figure 6: GENESIS-0 Core Lay out**

The GENESIS-0 core is fully representative of the ASTRID CFV BD core for what concerns the leakage component of the sodium void reactivity effect (SVRE). However, it is not the case for the central component of the SVRE. The ZONA4FW fuel cell uses platelets that have low content of Pu240 and hence the adjoint flux slope ($\xi$) is not very representative of the ASTRID Basic Design core one. Another cell has hence been defined using higher Pu240 content. But this one uses fuel pins which have a 30.6 cm long and hence the core GENESIS-1 (Figure 7) no longer exhibits a fully representative axial geometry. Hence this core will be better representative of the central component (Figure 8) but much less for the leakage component.

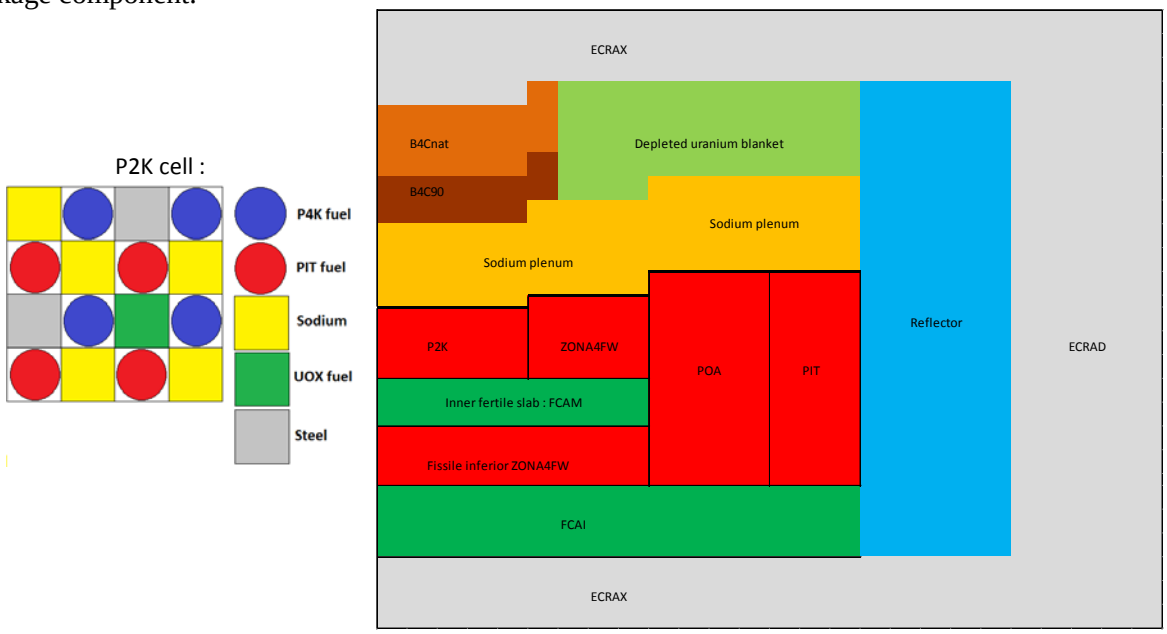


**Figure 7: GENESIS-1 Core Lay out**

Hence this core will be better representative of the central component (Figure 8) but much less for the leakage component.

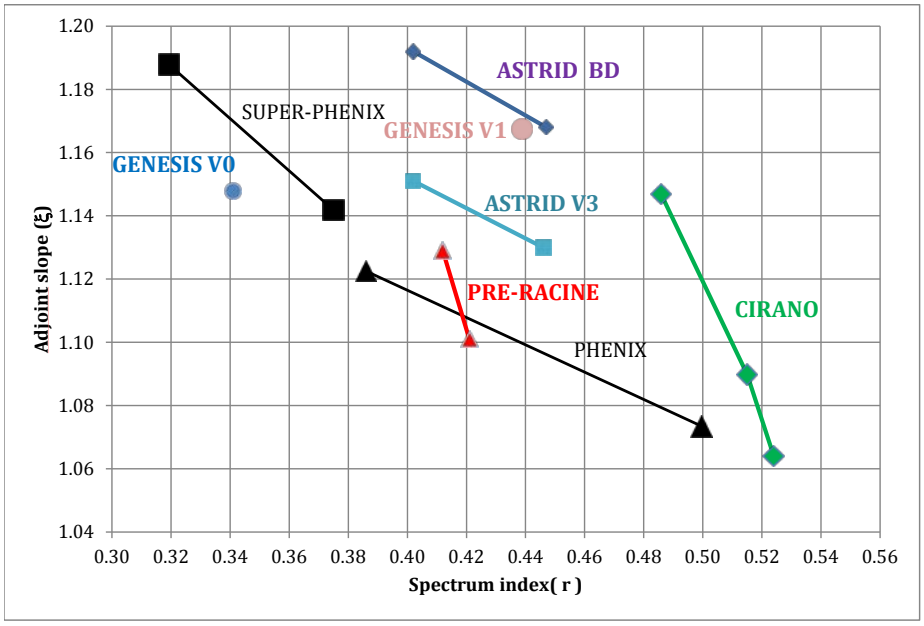


**Figure 8: Characteristics of the GENESIS Experimental cells**

Maxence's PhD work[20] has shown the link between the LC and the power distribution. It is why the measurement of the axial flux distribution through the use of different fission chambers.

## 6. CONCLUSIONS

Experiments are required for defining precisely characteristics of the ASTRID core (Advanced Sodium Technological Reactor for Industrial Demonstration).

Past experiments will allow a quantification of the uncertainties for core characteristics such as βeff or Doppler, a reduction of the uncertainties due to nuclear data (through assimilation) but will not cover all requirements. It will not cover for instance ASTRID core specificities such as sodium void reactivity, control rod worth, power map distribution over the cycle.

The MASURCA facility is currently under refurbishment until 2023 and will exhibit improved performances. The future GENESIS experimental programme of the MASURCA facility will take advantage of advanced measurement techniques, on one hand, and the possibility of using fuel with degraded Pu vector to develop a solid knowledge of ASTRID core characteristics. For addressing these particular issues, a core with the ZONA4-FW core cell using existing materials that can be used to faithfully represent the core geometry of the ASTRID CFV-BD. A variant of that core has been also designed using a cell with higher Pu240 content similar to the ASTRID CFV-BD core. This will enable a better representation of the central component of the sodium void reactivity effect (SVRE).

Independently from that, a better knowledge of the neutron balance of the plant is necessary either at BoC or EoC. This will be done by other types of measurements such as absolute fission chambers, noise techniques, open loop reactivity measurement techniques.

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