# ADVANCES IN FAST NEUTRON REACTOR NEUTRONICS CALCULATIONS

# AN OVERVIEW OF THE VALIDATION PROCESS USING MEASUREMENTS PERFORMED IN FAST REACTORS PHENIX AND SUPER-PHENIX

S. CZERNECKI, F. VARAINE and J.TOMMASI
CEA-Cadarache - Reactor Physics and Fuel Cycle Division,
Bldg230, 13108 Saint Paul Lez Durance Cedex, FRANCE
czernec@gaulois.cad.cea.fr ; fvaraine@cea.fr ; jtommasi@cea.fr

## ABSTRACT

The fast reactor european neutronics calculations system, ERANOS, has integrated recent improvements both in nuclear data, with the use of the adjusted nuclear library ERALIB1 from the JEF2.2 library, and calculation methods, with the use of the new european cell code, ECCO, and the deterministic code, TGV/VARIANT. This code performs full 3-D reactor calculation in the transport theory with variational method. A new calculational scheme for fast spectrum systems offering good compromise between accuracy and running time has been developed using these improvements plus a special procedure accounting for control rod heterogeneity, which uses a reactivity equivalence homogenization. The new scheme has been validated by means of experiment/calculation comparisons, using the extensive start-up program measurements performed in Super-Phénix reactor. The validation uses also recent measurements performed in Phénix reactor. The results are very satisfactory and show a significant improvement for almost all core parameters, especially for critical mass, control rod worth and radial subassembly power distribution.

## 1. INTRODUCTION

Fast reactors due to their superior neutron economy have a potentially important role to play in the future for an improved use of natural ressources or to burn effectively plutonium and minor actinides in the frame of the fuel cycle management back-end optimization. Moreover, new concepts are being explored and developed, such as accelerator-driven systems also based on the use of fast neutron cores in a subcritical mode. These new applications can need improved methods and data, in order to reduce uncertainties and to give confidence in their announced performances which often represents a delicate compromise (e.g. between burning capabilities, acceptable burn-up reactivity swing and reactivity coefficients). In recent years, important improvements have been achieved both in nuclear data and calculation methods. Partly, this was made possible by the constant increases in calculation capabilities. These improvements could potentially answer the physics requirements of fast spectrum systems for new applications.

Hence, a new design calculation scheme has been developed at CEA using improved nuclear data and calculational methods. This system is called ERANOS (European Reactor Analysis Optimized System) and is described in section 2. A new design calculation route obviously requires qualification over a specific domain of validity. The first step of the qualification process consisted in comparing calculation/experiment (C/E) values for the main neutronic core characteristics, such as critical mass, control rod worth and the power and flux distributions. The experimental values are taken from the extensive start-up program of measurements performed in the Super-Phénix [1] and Phénix reactor. They are introduced in section 3. During this qualification, the improvement of the performance of the new calculation scheme compared to the previous one was found to be noticeable for all the core neutronic parameters, but especially for the radial subassembly power distribution and the control rod worth, which were very poorly predicted by the old scheme [2,3]. The main features of the experiment/calculation and old/new scheme comparisons are summarized in section 4.

## 2. A NEW DESIGN CALCULATION ROUTE

The new design calculation scheme is based on deterministic methods. It has been developed in release 1.2 of the ERANOS code and data system and consists of :

1. ECCO [5], the European Cell Code. ECCO prepares self-shielded cross sections and scattering matrices for every core material using both the subgroup method within each fine group and the slowing down treatment in many groups (1968 groups). ECCO can even treat explicitly complex heterogeneous geometries [6] like 2D hexagonal geometries when calculating the collision probabilities. The flux and current are calculated in a P1 approximation:

$$\Phi_i^g = \sum_{j=1}^{n} P_{\mathrm{ji}}{}^g (S_j^g - B^g J_j^g)$$
$$J_i^g = \sum_{j=1}^{n} P_{\mathrm{ji}}{}^g \left(\sum_{g'} \Sigma_{s1,j}^{g'\to g} J_j^{g'} + \frac{B^g}{3}\Phi_j^g\right) \quad (1)$$

$$S_j^g = \sum_{g'\le g} \Sigma_{s0,j}^{g'\to g} \Phi_j^{g'} + S_{fj}^g \quad (2)$$

where n = total number of regions
$S^g_{fj}$ = the fission source in the group g
$J^g_j$ = current in group g in region j
$\Phi^g_j$ = flux in group g in region j
$\Sigma_{s0,j}^{g'\to g}$ , $\Sigma_{s1,j}^{g'\to g}$ = order 0 and order 1 scattering cross section into group g in region j
$B^g$ = buckling in group g
$P_{i,j}$ = reduced collision probabilities in region j for a neutron born in region i (i.e. divided by the total cross section).

2. the TGV/VARIANT code [7]. The TGV/VARIANT code performs three-dimensional (hexagonal-Z) nodal transport calculations. TGV/VARIANT was employed using $6^{th}$ order polynomial approximation inside the node, flat leakage approximation on the surface and simplified spherical harmonics [8] approximation for the angular dependence, with 33 energy groups. TGV/VARIANT allows to get correction free from mesh and transport [8].

3. a procedure for creating homogeneous cross sections for control rods. This procedure uses a reactivity equivalence method [9,10] with the SN transport code BISTRO [11] and its associated perturbation modules to generate an homogenized set of the absorber cross sections for use in transport theory. The condition of equivalence is derived by imposing the reactivity variation $\Delta\rho_{het}$, between the heterogeneous configuration and the homogeneous equivalent one, equal to zero. Using exact perturbation theory, we obtain:

$$\Delta r_{het} = \iint \Psi_{het} (\Sigma_{\mathrm{hom}} - \Sigma_{het}) \Psi^*_{\mathrm{hom}} = 0 \quad (3)$$

It leads to:

$$\Sigma_{\text{hom}}^{g} = \frac{\iint \Psi_{het}^{g} \Sigma_{het}^{g} \Psi_{\text{hom}}^{*g}}{\iint \Psi_{het}^{g} \Psi_{\text{hom}}^{*g}} \quad (4)$$

where $\Psi_{\text{het}}$ denotes the angular flux calculated for the heterogeneous structure cell
$\Psi_{\text{hom}}^{*}$ is the adjoint flux calculated using the homogenized cross-sections
$\iint$ stands for spatial and angular integration
g denotes the group energy.

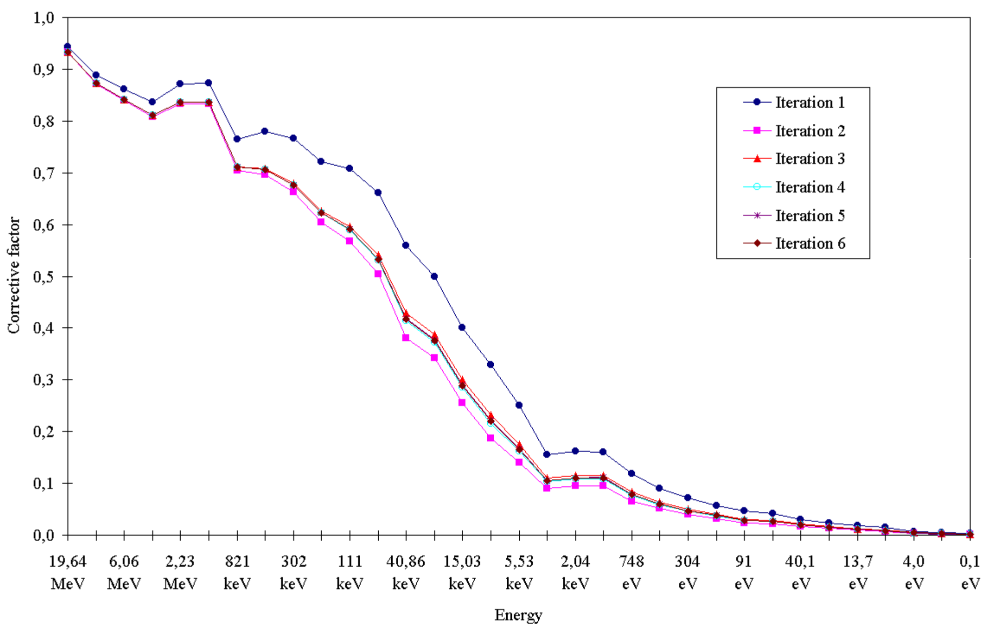
For determining scattering cross sections from a group g into g' (g ≠ g'), $\Psi_{\text{hom}}^{*g}$ is replaced in the equation (4) by $\Psi_{\text{hom}}^{*g'}$. And for self-scattering cross section from a group g, $\iint \Psi_{het}^{g} \Psi_{\text{hom}}^{*g}$ is replaced by $\iint \Psi_{het}^{g} \Psi_{\text{hom}}^{*g} - \int \Phi_{het}^{g} \int \Phi_{\text{hom}}^{g*}$, where $\Phi$ is the scalar flux obtained by angular integration. $\Psi_{\text{hom}}^{*}$ is unknown. An iterative procedure must be followed in which the homogenized cross-sections used to obtain the adjoint flux solutions for the homogenized model are improved in successive cycles. In the first cycle the adjoint flux is calculated using flux x volume averaged homogenized cross-sections. We define for each reaction types, corrective factors $f_g$ as:

$$f_{x,g}^{(i)} = \frac{\Sigma_{\text{hom}}^{g,(i)}}{\Sigma_{vol}^{g}}$$

where x stands for reaction type
g stands for energy group
$\Sigma^{(i)}_{\text{hom}}$ is the equivalent homogenized cross-section calculated at the loop i of the iterative procedure
$\Sigma_{\text{vol}}$ is the volume homogenized cross-section.

The Figure 1 illustrates the iterative procedure for the capture cross-section. After 3 iterations the procedure has nearly converged. The good convergence of the procedure in every cases must be underlined.

Figure 1. Corrective factors on the capture cross section



In addition to the refined calculation methods, improvements in nuclear data are included in the new scheme. This has been achieved by using the recent JEF2 [12] evaluation, which is further improved by using nuclear data adjustment procedures. The experimental basis for the adjustment is made up of 350 clean core critical experiments, and the resulting adjusted library is called ERALIB1 [13].

In order to highlight the improvements brought up by the new design scheme, we briefly recall that the old scheme was also based on deterministic methods:

1. nuclear data were taken from the adjusted CARNAVAL-IV data set [14].
2. the self-shielded cross section sets were prepared using the HETAIRE cell code [14].
3. the modular CCRR code scheme [15] was used to perform all calculations. The flux calculation is derived in a three dimension geometry (Hexagonal-Z) with a radial mesh of seven points per hexagon, with 6 energy groups, using the diffusion theory approximation to the neutron transport equation.

## 3. EXPERIMENTAL INFORMATION

As indicated previously, the experimental information is drawn from nuclear power plant measurements. At the start-up of the Super-Phénix reactor, an extensive experimental program has been built-up in order to verify the design tool performances and to collect data pertaining to future fast reactors research and development. From this point of view, the Super-Phénix start-up tests have been a unique source of information for the assessment and validation of neutronics data and methods The main core characteristics are summarized in table I and in Figure 1. Detailed characteristics are gathered in the special volume of Nuclear Science and Engineering entirely devoted to the Super-Phénix reactor (Vol 106), edited in September 1990 [1].

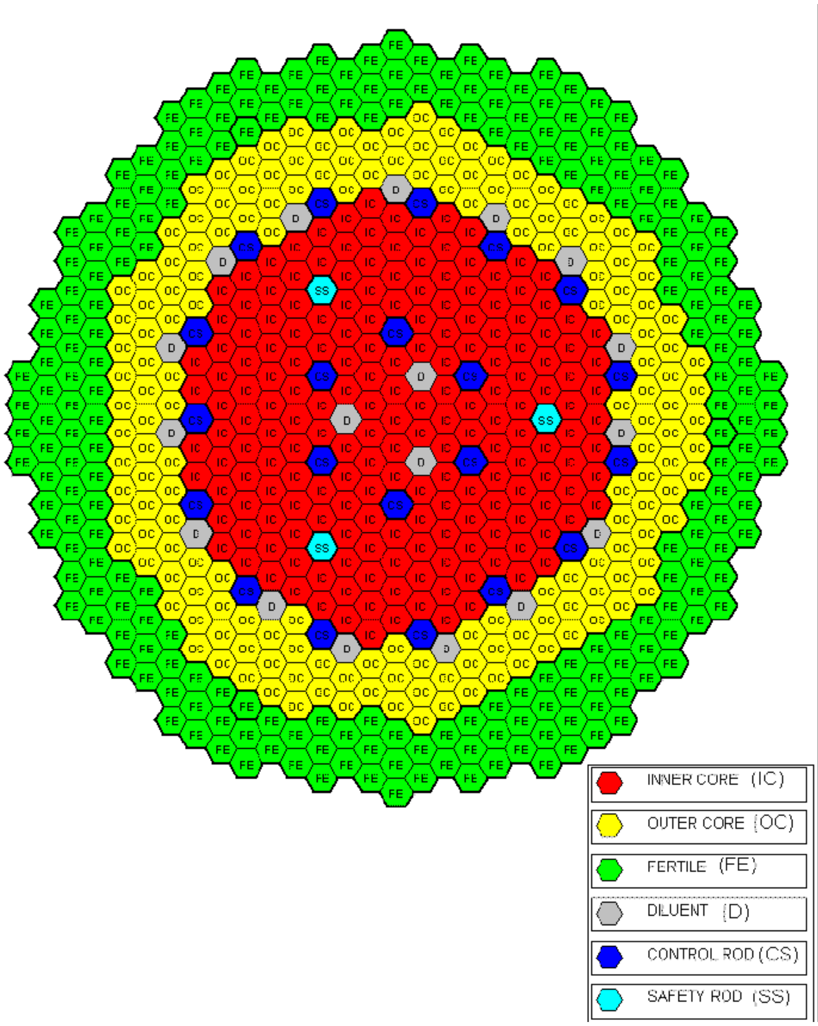
During the commissioning test of Super-Phénix, most main neutronic core characteristics have been measured, such as critical mass and control rod worth [2], the power and flux distributions [3,4], reactivity and feedback distributions [5].

Table I. Super-Phénix and Phénix core characteristics

| | Super-Phénix | Phénix |
|---|---|---|
| core volume (liter) | 10800 | 1200 |
| Fuel | Sintered $UO_2$-$PuO_2$ | Sintered $UO_2$-$PuO_2$ |
| Thermal power rating (MWth) | 3000 | 563 |
| Number of fissile/fertile | 376/233 | 121/90 |
| Number of primary control rods | 21 | 6 |
| Plutonium enrichment of the fuel (weight fraction %) :<br>Inner zone<br>Outer zone | <br>15<br>18 | <br>18<br>25 |
| Plutonium mass (kg) | 5600 | 800 |
| Breeding gain | 0.24 | 0.13 |
| Maximum flux (neutrons/$cm^2$s) | $6\ 10^{15}$ | $7.2\ 10^{15}$ |
| Sodium temperature (°C)<br>Core inlet/outlet | <br>395/545 | <br>382/532 |

Starting from the fact that the commissioning tests of Super-Phénix have provided reliable and validate measurements methods for almost all core parameters, experimental information is also drawn from the Phénix reactor. Some core characteristics are summarized in table I. In fact, a recent measurements program, called REACTIVIX, took place in the Phénix reactor. It consists in updating the control rod worth of Phénix by using a reference measurement method: the static source multiplication method (MSM) instead of balancing of two rods with the reactor critical.

Figure 2. Super-Phénix core layout




## 4. QUALIFICATION PROCESS

The qualification process has consisted in comparing calculation/experiment values. Where no experimental values are available, the comparison has been performed with respect to reference calculations but most of the main neutronic core characteristics has been validated against experiment. Special care has been taken on assessing parameters which were poorly calculated in the past. It was particularly the case for the radial power distribution and the control rod reactivity worth. These comparisons are presented here.

### 4.1 CRITICAL STATES AND CONTROL ROD WORTH

The program of control rod experiments at the start-up of Super-Phénix has been very extensive. Plenty of core situations have been measured. It includes critical states with different control rod positions and many control rod worth. Table II gives the comparison between experiment and calculation for control rod worth and Table III gives the main results in reactivity measurements.

For control rod worth, the comparison is well within experimental uncertainty, and compared to the previous C/E obtained with the old scheme, the results are impressive. Moreover, the previous results were obtained after a lot of method corrections (coarse mesh, transport and heterogeneity) had been determined and applied whereas the new calculated values are directly obtained.

Table II. Comparison of calculation to measurement in Super-Phénix control rod worth measurements

| Control rod configuration | E ($10^{-5}$ Δk/k) | C ($10^{-5}$ Δk/k) | **C/E** | C/E (old scheme) |
|---|---|---|---|---|
| Δρ(SCP[1]), SAC[2] ↑[3] | 8067 ± 995 | 8119 | **1.006** | 1.198 |
| Δρ(SAC), SCP ↓ | 1193 ± 155 | 1115 | **0.935** | 1.205 |
| Δρ(SAC), SCP at critical height | 1039 ± 120 | 1009 | **0.971** | 1.102 |
| Δρ(Internal triplet), SCP at height 465 mm | 1530 ± 184 | 1505 | **0.984** | 1.080 |
| Δρ(External triplet), SCP at height 486 mm | 1115 ± 134 | 1106 | **0.992** | 1.053 |

[1] SCP refers to the main control rod system
[2] SAC refers to the safety rod system
[3] Arrows refers to control rod down (↓) or up (↑)

For critical states and reactivity measurements, the agreement is also satisfactory. Given that control rod worth are precisely predicted, critical mass and interaction effects with control rod are correctly calculated. Note that new values are also directly obtained. Although the previous scheme results were satisfactory [2], they were obtained after many additional corrections, even a 20% correction to control rod worth deduced from calculation/experiment comparisons (Table II).

Table III. Summary of the main C-E observed in reactivity measurements

| | E ($10^{-5}$ Δk/k) | C ($10^{-5}$ Δk/k) | **C-E ($10^{-5}$ Dk/k)** |
|---|---|---|---|
| Full power core (CMP), SCP ↑ | 3710 | 3745 | **+35** |
| CMP, SCP at critical height | Critical states (E = 0) | +70 | **+70** |
| CMP, internal curtain of rods down, external curtain of rods at height 661 mm | | +56 | **+56** |
| CMP, internal curtain of rods at height 964 mm, external curtain of rods down | | -24 | **-24** |

In order to extend the qualification basis, the control rod worth experiment in Phénix has been used. In the Super-Phénix way, many control rod worth have been measured. Table IV gives the main results. They are quite good as well. The improvement of the C/E values compared to the old scheme is noticeable. The results are obviously consistent with Super-Phénix one’s.

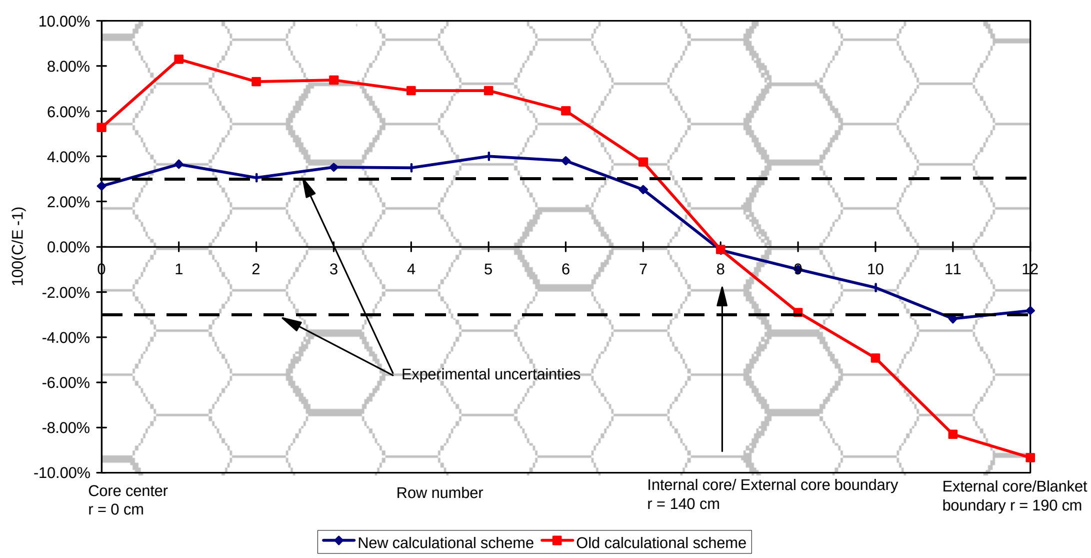
Table IV. Summary of the main results observed in Phénix control rod worth measurements

| | Experimental value ($) | Calculation ($10^{-5}$ Δk/k) | **C/E (1$ = 0.339% Dk/k))** | C/E (old scheme) |
|---|---|---|---|---|
| Main control rod system reactivity worth | 23.28 ± 1.06 | 7752 | **0.98** | 1.092 |
| Reactivity worth of three individual rods | 3.54 ± 0.18<br>3.48 ± 0.18<br>3.74 ± 0.19 | 1205<br>1190<br>1251 | **1.00**<br>**1.01**<br>**0.99** | 1.152<br>1.145<br>1.108 |
| Partial reactivity worth of individual rods<br><br>(Δρ = ρ(6 rods at height 608 mm) - ρ(5 rods at height 608 mm, one is down) | 2.69 ± 0.15<br>2.51 ± 0.14<br>2.58 ± 0.14<br>2.66 ± 0.15<br>2.75 ± 0.15<br>2.86 ± 0.16 | 936<br>895<br>907<br>925<br>944<br>972 | **1.03**<br>**1.05**<br>**1.04**<br>**1.03**<br>**1.01**<br>**1.00** | 1.160<br>1.215<br>1.195<br>1.151<br>1.130<br>1.111 |
| Partial reactivity worth of the control rod system<br>(Δρ = ρ(6 rods at height 608 mm) - ρ(6 rods down) | 18.29 ± 0.42 | 6090 | **0.98** | 1.112 |

## 4.2 RADIAL POWER DISTRIBUTION

The comparison between calculation and experimental results has been performed on the first thermal balance at full power in November 1986. In the Figure 3, the deviations between calculation and experimental values are given in percent. This figure emphasizes the important improvement brought by the new calculation scheme for the prediction of the power map. The old calculations show a 7% overestimation at the center and an underestimation of that quantity by up to 10% for the periphery. The new calculations show a 3% overestimation at the center and an underestimation of that quantity by only 3% for the periphery. It is evident that the old calculated values are outside the deviation allowed by experimental uncertainties although the new calculated values are nearly inside.

Figure 3. Row averaged relative error (in percent) for the power per subassembly between calculation and measurement in Super-Phénix.



Several points have been investigated in order to understand the differences between the calculated powers by the old and the new scheme:

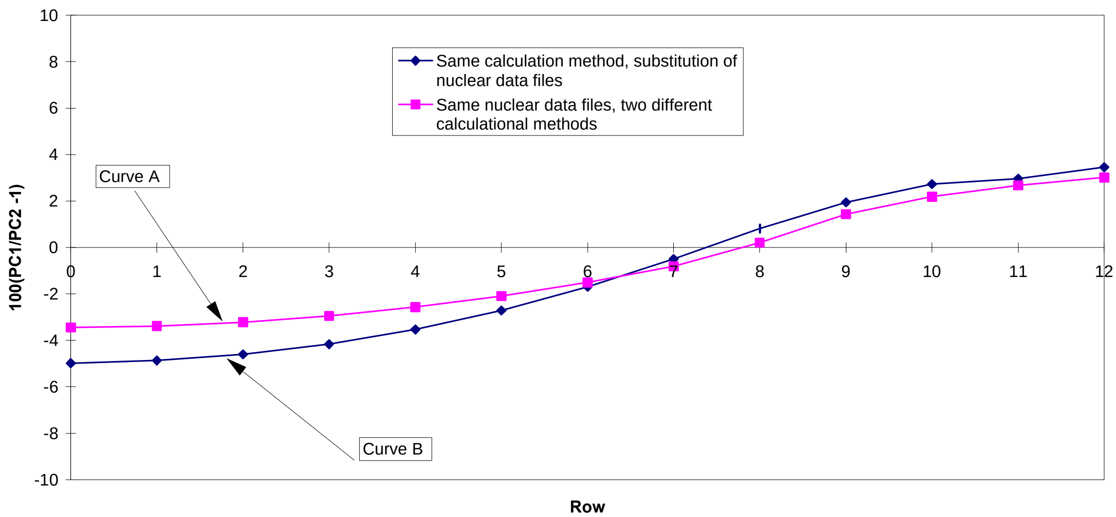
- the method effects
- the influence of the nuclear cross section data sets

Previous studies have already investigated the method influences on the radial power distribution [3]. It was found that the mesh effect and the transport effect are the most important one. Taken together, these effects mean that the old scheme methods (coarse mesh and diffusion) leads to an overestimation of the ratio of the flux at the core center to that on the periphery by 4%. This is confirmed by this study as shown by the Figure 4 (Curve A). It represents the relative difference, averaged by row between powers by subassembly calculated using identical data sets and:

1. with TGV/VARIANT, simplified transport theory, 33 energy groups
2. with a finite differences diffusion calculation, 6 energy groups, 7 point radial mesh in a subassembly

This relative difference is about 4%. It means that old standard calculational methods effects are explicitly taken into account by a TGV/VARIANT calculation. This is due on one side to the possibility of using transport theory calculation for a whole core in a Hexagonal-Z geometry, on the other side to negligible mesh effect of variational methods compared to finite differences methods [8].

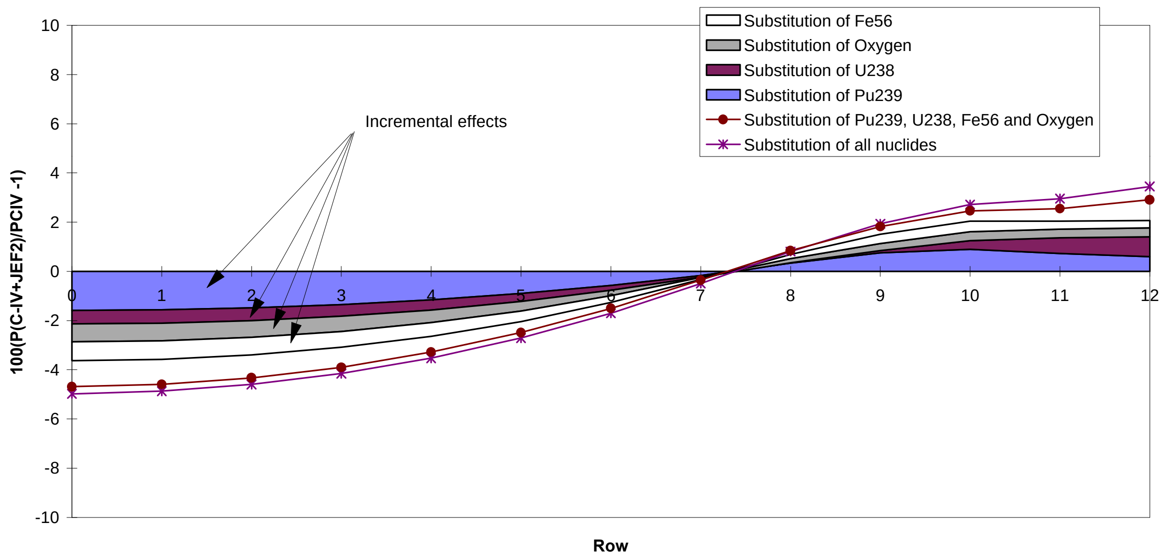
Figure 4. Row averaged relative differences between the power per subassembly calculated by two different ways.




The influence of the nuclear cross section data sets has also been studied. Globally, the change of the data sets using identical methods decreases the radial gradient by a value of 6% as shown in Fig. 3 (Curve B). This effect, which is the most important one, has been examined more deeply. The nuclides which contribute to this effect have been identified. In order to identify them, a substitution of nuclides has been performed in the CARNAVAL-IV set. So that, one by one, CARNAVAL-IV cross sections have been replaced by those of JEF2. Then, the power per subassembly is compared to a full CARNAVAL-IV calculation. The results show that only four nuclides have noticeable contributions to the data set effects which are Pu239, U238, Fe56 and Oxygen (Figure 5). If the four nuclides are substituted, the global effect is obtained (Figure 5).

The experiments involved in the qualification basis included burn-up swing, Doppler effect and also an extensive analysis of flux distributions within the core obtained by irradiation of fission and activation foils and by fission chambers measurements. The results obtained are also very satisfactory [16].

Figure 5. Row averaged relative differences between the power per subassembly calculated with CARNAVAL-IV library with the data for one nuclide substituted by JEF2 data and the full CARNAVAL-IV library, at full power state, in heterogeneous geometry.




## 5. CONCLUSION

The results obtained with the new calculation tools on Super-Phénix start-up experiments are very impressive. The extension of the qualification basis to Phénix has been performed and the results are satisfactory as well. The values of the neutronic core parameters are obtained without applying any post-correction whereas a lot of such corrections were required with the older scheme. This point makes the new scheme use simpler.

This scheme provides reliable means for neutronics studies of fast spectrum new applications. Its high accuracy and modelization capabilities has already been used for the neutronic studies carried on the fast reactor Phénix.

## ACKNOWLEDGEMENTS

We would like to thank all the colleagues from CEA who helped us during this work.

## REFERENCES


1. J. GOURDON, B. MESNAGE and al., “An overview of Super-Phénix commissioning tests,’’ *Nuclear Science and Engineering,* **106**, pp. 1-10 (1990).

2. J.C. GAUTHIER and al., “Measurements and predictions of control rod worth,’’ *Nuclear Science and Engineering,* **106**, pp. 18-29 (1990).

3. J.C. CABRILLAT and al., “Power and neutron flux distributions in the core and shielding, ’’ *Nuclear Science and Engineering,* **106**, pp. 37-46 (1990).

4. D. LETEINTURIER, B. BERNARDIN and al., “Measurement of core thermal-hydraulic characteristics, ’’ *Nuclear Science and Engineering,* **106**, pp. 47-58 (1990).

5. S. RAHLFS, Doctoral thesis in science, University of Marseille, FRANCE, 1995.

6. G. RIMPAULT, M.J. GRIMSTONE, “Validation of new sub group algorithms for resonance self shielding in heterogeneous structures, ’’ Topical Meeting on Advances in Nuclear Computation and Radiation Shielding, SANTA FE NEW MEXICO, (April 1989).

7. G. PALMIOTTI, C.B. CARRICO and E.E. LEWIS, VARIANT : VARIational Anisotropic Nodal Transport for Multidimensional Cartesian and Hexagonal Geometry Calculation, ANL-95/40, Argonne National Laboratory (1995).

8. E.E. LEWIS, G. PALMIOTTI, “Simplified Spherical Harmonics in the Variational Nodal Method, ’’ *Nuclear Science and Engineering,* **126**, 48-58 (1997).

9. J.L. ROWLANDS, “Physics of fast reactors control rods, ’’ *Progress in Nuclear Science,* **16**, pp. 287-321 (1985).

10. J.L. ROWLANDS, C.R. EATON, “The Spatial Averaging of cross sections for use in Transport Theory Reactor Calculations, with an application to Control rod fine structure homogenization, ’’ Proc. Int. Mtg. on Homogenization methods in Reactor Physics. LUGANO SWITZERLAND, (1978).

11. G. PALMIOTTI et al., “Optimized two dimension SN Transport code (BISTRO), ’’ *Nuclear Science and Engineering,* **104**, pp. 37-46 (1990).

12. E. FORT, M. SALVATORES, “JEF2 Validation,’’ Proceedings Inter. Conf. on nuclear data for science and technology, Gatlinburg USA, (1994).

13. E. FORT and al., “Realization and performance of the Adjusted nuclear data library ERALIB1 for calculating fast reactor neutronics, ’’ Proceedings of PHYSOR 96, MITO JAPAN, (September 1996).

14. J.P. CHAUDAT et al., “The CARNAVAL-IV adjusted library, ’’ *Trans. AM. Nuclear society,* **27**, 877 (1977).

15. G. GIACOMETTI et al., “CCRR, ’’ Proceedings Topl. Mtg. Advances in Nuclear Engineering Computational Methods, KNOXVILLE USA, (April 1985).

16. S. CZERNECKI, Doctoral thesis in science, University of Marseille, FRANCE, (1998).