ASSESSMENT OF LATEST DEVELOPMENTS IN SODIUM VOID REACTIVITY WORTH CALCULATIONS

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ABSTRACT

A renewed assessment of the sodium void reactivity worth is being presented in this paper after new cross-section evaluations (JEF2)[1] and computational methods (ECCO/ERANOS)[2,3] have been established. The CIRANO[4] experimental programme, which provides the experimental data base for Plutonium Burning cores[5] (CAPRA design) is also a reason for such a renewed study.

A detailed analysis of the experiments requires the use of transport theory with an accurate representation of anisotropic streaming and the use of perturbation theory for transport methods. This allows the separation of the global effect into a leakage term and a non-leakage term and the application of known method corrections (the anisotropy of streaming not yet implemented) to the determination of the leakage components.

The discrepancies observed for the negative leakage term are important using the current computational scheme but it has been shown that it is caused by the existence of a strong anisotropy of streaming due to existing leakage corridors in the MASURCA lattice.

The positive non-leakage term is sensitive to nuclear data and its prediction presents some small discrepancies in comparison with the experimental results. The sodium void reactivity worth in plutonium burning cores has a different sensitivity to nuclear data and the analysis of the CIRANO programme has shown that this can be correctly predicted.

INTRODUCTION

Sodium void reactivity worth is an important safety parameter to be considered for the design of fast reactors. Its determination requires both refined nuclear data with reduced uncertainties and improved calculation methods. This has been recognised for classical fast

reactor designs (breeders), and is now increasingly important for new plutonium and minor actinide burning cores such as those of the CAPRA project[5].

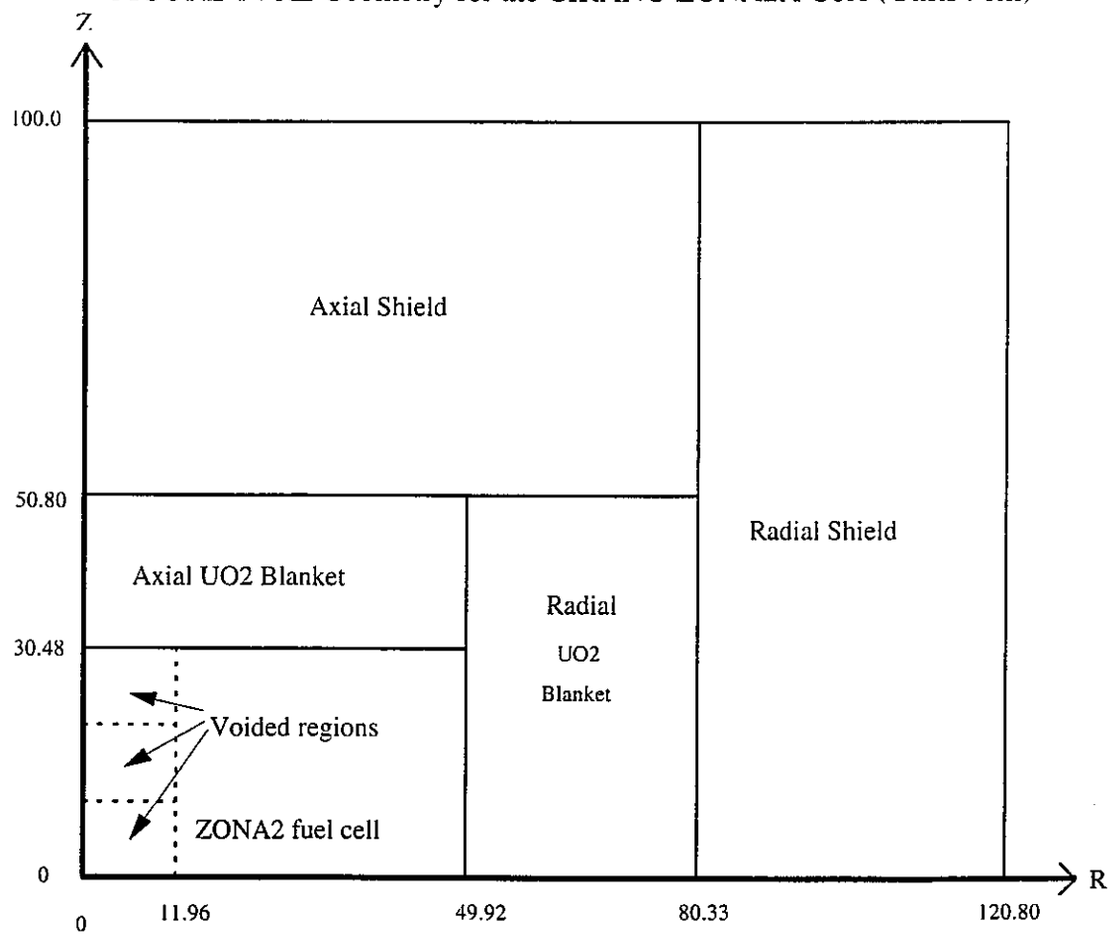
There are several reasons for a renewed neutronic assessment of the sodium void reactivity worth :

- new nuclear data evaluations (for example JEF2) have been produced,
- deterministic methods have been improved (for example the cell code ECCO[2] and the ERANOS[3] code system),
- new experimental measurements are available (for example the CIRANO[4] programme).

EXPERIMENTS

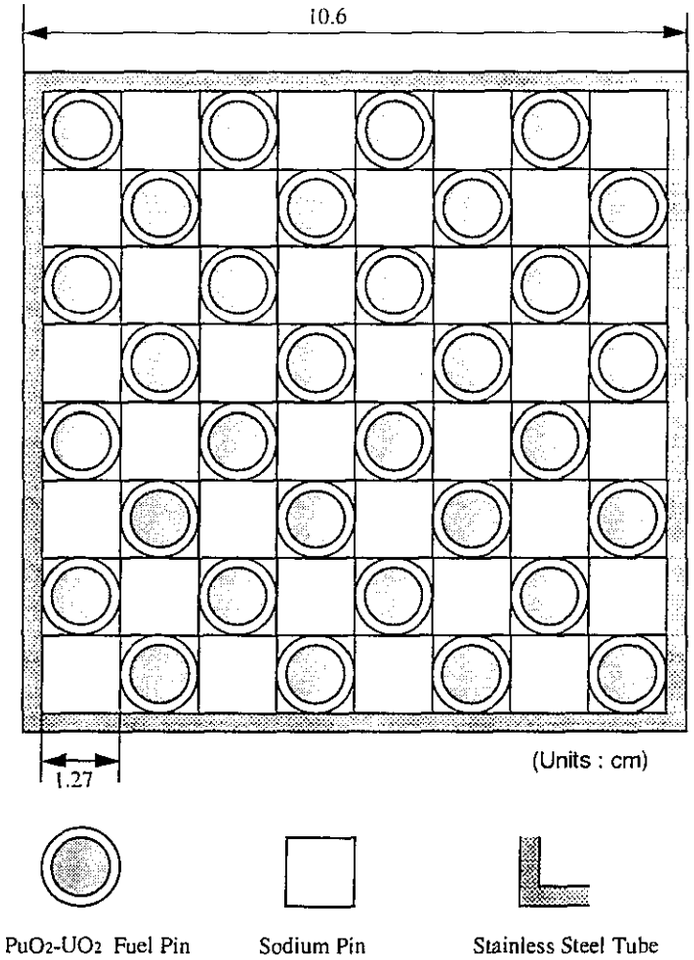
The CIRANO experimental programme is underway at the MASURCA critical facility with the purpose of investigating the characteristics of plutonium burning fast reactors. During the first phase of the CIRANO programme the sodium void worth was measured in the central four assemblies of the CIRANO ZONA2A core. This core has been established in such a way that RZ modelling is very representative (Figure 1). The core contains a unique ZONA2 PuO_2 -UO_2 fuel cell with a plutonium enrichment of around 25% (Figure 2).

FIGURE 1 : RZ Geometry for the CIRANO ZONA2A Core (Units : cm)



The sodium void reactivity worth is measured when replacing sodium pins by voided ones in the central core zone.

FIGURE 2 : Two-Dimensional Model of the ZONA2 PIT Cell



Four different measurements were performed over several axial heights in order to obtain a sodium void reactivity worth with different leakage components. For the smaller reactivity worth measurements, the reactivity worth was determined using the reactor control rod calibrated using inverse kinetics, while for the larger worth measurements counters were used with the reactor in a sub critical state. Similar measurements were performed in sodium/steel reflected cores and have shown no major change in the sodium void reactivity worth. Hence the effect of replacing the axial and radial UO_2 fertile blankets by sodium/steel reflector elements does not seem to influence the sodium reactivity worth in the core.

To obtain a comparison with a more conventional fast reactor configuration an analysis was performed for the ZONA1/R1 configuration of the PRE-RACINE[6] programme. The core (Figure 3) contains a ZONA1 PuO_2 -UO_2 fuel cell with a plutonium enrichment of around 18% (Figure 4) surrounded by a R1 uranium fuel cell of similar spectrum with a U235 enrichment of around 30%. Two similar cores were established using plutonium oxide fuel with a different Pu240 content (8 % and 18 %) in the central region. As for the CIRANO ZONA2A configuration, five different measurements were performed over a selection of axial heights. The experimental uncertainties are given to be ±5% for the central void, and are of similar absolute magnitude for the other void measurements.

FIGURE 3 : RZ Geometry for the ZONA1/R1 PRE-RACINE Clean Core (Units : cm)

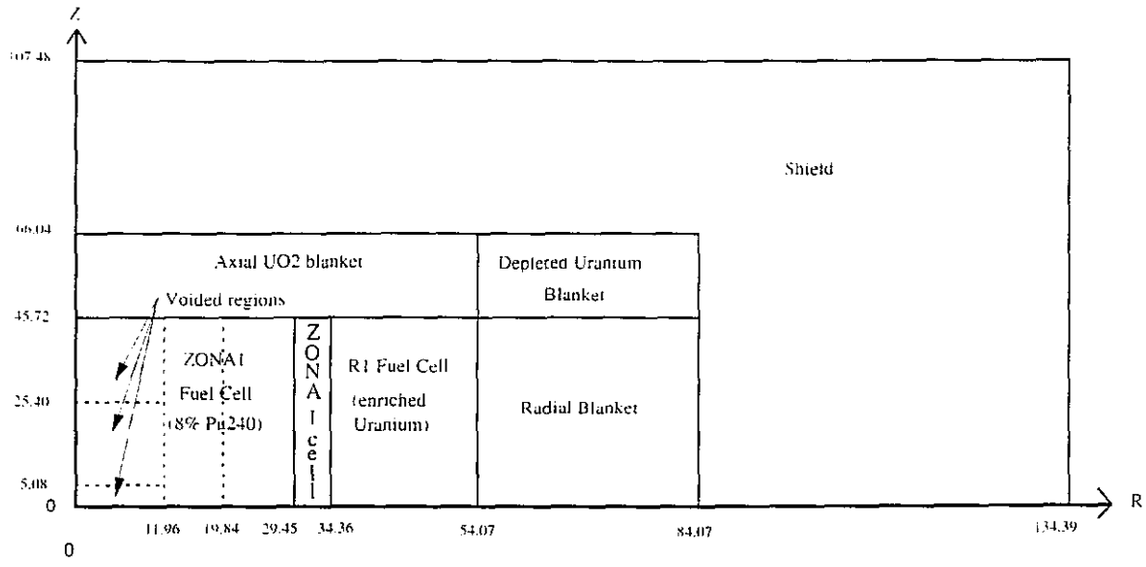
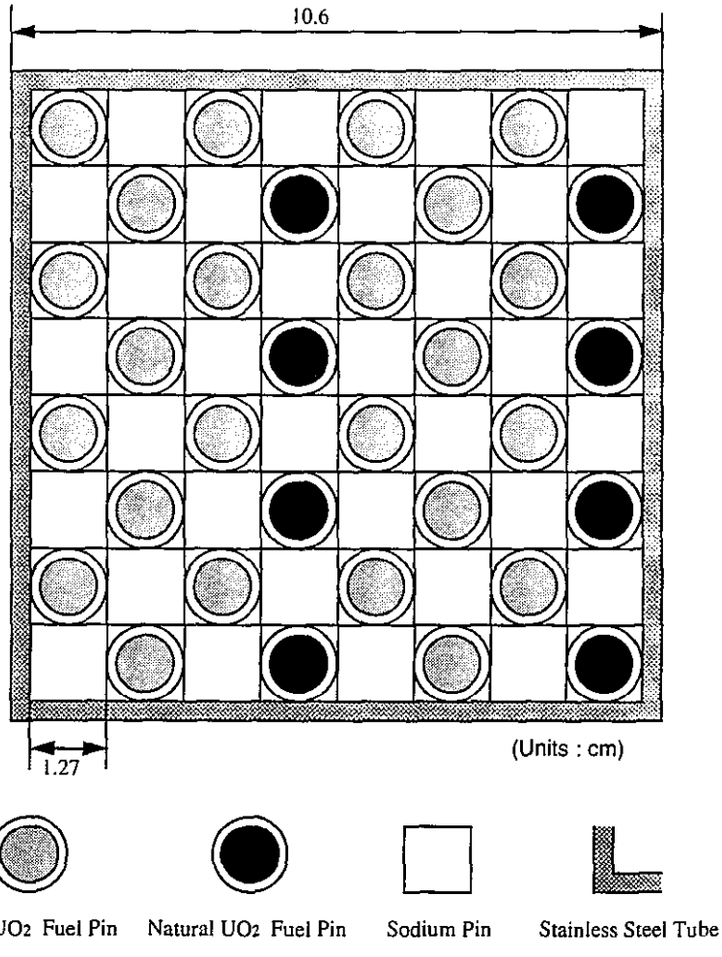


FIGURE 4 : Two-Dimensional Model of the ZONA1 Cell



CALCULATION SCHEME

The calculation scheme used for this analysis has been developed within the European collaboration on Fast Reactors. It consists of using the JEF2 evaluated data library, the ECCO cell code and the ERANOS code system.

The ECCO[11] cell code uses the subgroup method to treat the resonance self-shielding, this being particularly suitable for calculations involving complex heterogeneous structures. Libraries have been generated[7] from JEF2 evaluations with the processing codes NJOY[8] and CALENDF[9] for use in ECCO, and contain sub-group parameters to represent the resonance structure of the cross-sections. An explicit P1 leakage treatment with an appropriate self-shielding determination has been introduced in ECCO, therefore avoiding the use of the transport cross-section. A simplified but accurate treatment of the fertile blankets is performed taking into account the neutron source from the core[16].

Effective cross sections and matrices are then used in Diffusion, SN P0 and SN P1 Transport calculations with the BISTRO module of the ERANOS code system. For the experimental analysis, perturbation theory calculations have been performed within ERANOS using the different options listed above.

ECCO prepares self shielded cross sections and matrices using both the subgroup method within each fine energy group and the slowing down treatment in many energy groups (1968 groups). The ECCO reference route treats the heterogeneous geometry in fine groups (1968) for the 37 most important nuclides. Less important nuclides are taken from the broad group libraries (172 or 33 groups). These calculations are very accurate as the fine group plus sub-group scheme has been set up to represent accurately the reaction thresholds and narrow or wide resonances.

The self shielding formula for a standard effective cross section $_x$, where x can be the total (Legendre order 0), capture, fission, elastic or inelastic reaction for each group g, is as follows:

$$\tilde{\sigma}^g_{xi} = \frac{\sum_j S^g_j \sum_k \alpha^g_k \, \sigma^g_{x_k} \, p_{ij}\left(\Sigma^g_{t_k}\right)}{\sum_j S^g_j \sum_k \alpha^g_k \, p_{ij}\left(\Sigma^g_{t_k}\right)}$$

where S^g_j is the source in group g and in region j.

α^g_k is the probability in group g to obtain the partial cross section $\sigma^g_{x_k}$

to which corresponds the total cross section $\sigma^g_{t_x}$

(used to calculate the macroscopic cross section $\Sigma^g_{t_k}$ in each region).

$p_{ij}\left(\Sigma^g_{t_k}\right)$ is the reduced collision probability for the subgroup k within the group g.

The self shielding of the total Legendre order one cross section (as well as order one of the elastic cross section or the transport cross section if existing) has a different formulation arising from the fact that this type of cross section has a current weighting :

$$\tilde{\sigma}_{tr_i} = \frac{\sum_j S_j \sum_k \alpha_k \, \sigma_{tr_k} \sum_l p_{il}\left(\Sigma_{t_k}\right) p_{lj}\left(\Sigma_{t_k}\right)}{\sum_j S_j \sum_k \alpha_k \sum_l p_{il}\left(\Sigma_{t_k}\right) p_{lj}\left(\Sigma_{t_k}\right)}$$

The use of such a formulation, instead of the flux weighted one, has a significant impact on the leakage component. In fast reactors, oxygen resonances at high energies are responsible for this effect (a few hundred pcm for the critical mass determination).

The flux and current are calculated with the self shielded cross sections and matrices in a P1 consistent approximation :

$$\phi_i^g = \sum_j \left(-B^g \, J_j^g + S_{fj}^g + \sum_{g'} \Sigma_{s_{0,j}}^{g' \to g} \, \phi_j^{g'} \right) p_{ji}\left(\Sigma_t^g\right)$$

$$J_i^g = \sum_j \left((B^g/3) \, \phi_j^g + \sum_{g'} \Sigma_{s_{1,j}}^{g' \to g} \, J_j^{g'} \right) p_{ji}\left(\Sigma_t^g\right)$$

The effective cross sections and matrices are then condensed and homogenised to provide effective cross sections and matrices in the user required group scheme.

Finally, the fact that the neutronic balance is preserved in ECCO after condensation and homogenisation provides a final internal validation of the ECCO algorithms.

The ERANOS code system makes use of the effective cross sections produced by the ECCO cell code to perform full reactor calculations. In this study, the following ERANOS options were chosen :

- diffusion theory,
- Sn transport theory based on the S4 P0 (P1 corrected) formulation,
- Sn transport theory based on the S4 P1 formulation.

Also, calculations have been performed with the TGV transport code[14] based on the variational nodal method. This method enables the calculation of 3D rectangular and hexagonal geometries and for the purpose of this work gives the same results as the corresponding transport SN method. Full use of this method will be possible with the inclusion of perturbation theory modules that are currently being developed.

The sodium void reactivity worth consists of two basic components: a positive non-leakage term and a negative leakage term. To determine these two components perturbation calculations were performed using the ERANOS code system with both finite difference diffusion theory and transport theory (S4 P0 and S4 P1) approximations. The separation of the leakage and the non-leakage components of the reactivity worth in transport theory was performed in the following way using standard notations:

$$\text{Leakage term} = \sum_g \delta\Sigma_{t,g}\left\{\frac{1}{(4\pi)^2}\int_{4\pi} d\bar{\Omega}\Phi_g(\bar{\Omega})\int_{4\pi} d\bar{\Omega}\Phi_g^+(\bar{\Omega}) - \frac{1}{4\pi}\int_{4\pi} d\bar{\Omega}\Phi_g(\bar{\Omega})\Phi_g^+(\bar{\Omega})\right\}$$

$$+ \sum_g \sum_{g'} \sum_{\ell>1} \delta\Sigma_{s,g\to g',\ell}\ \phi_{g,\ell}$$

where $\Phi_g(\bar{\Omega})$ is the angular flux and $\phi_{g,\ell} = \int_{4\pi} d\bar{\Omega}\Phi_g(\bar{\Omega})\, P_\ell(\bar{\Omega})$.

Non-leakage term = Total reactivity worth - leakage term

RESULTS AND DISCUSSION

The results of both the direct and perturbation theory calculations have shown differences of less than 1.4 pcm for diffusion theory, and of less than 8.0 pcm for transport theory. The results obtained from this analysis have shown that the leakage and the non-leakage terms are in the following order :

Transport S4 P0 > Transport S4 P1 > Diffusion

The experimental values ρ_{exp_i} can be separated into non-leakage and leakage terms $\rho_{exp_i} = \alpha NL_{exp_i} + \beta L_{exp_i}$. The two components NL_{exp_i} and L_{cal_i} are not measured directly. However, C/E values have been deduced for the two sodium void worth components in each of the experimental configurations by the following method. The experimental reactivity was determined as $\rho_{exp_i} = \alpha NL_{cal_i} + \beta L_{cal_i}$, where NL_{cal_i}, L_{cal_i} are the calculated non-leakage and leakage terms, α and β, the bias factors for the non-leakage and leakage term respectively. The bias factors α and β were determined by minimising in least squares sense the calculation to measurement discrepancy. The function to be minimised can be defined as :

$$F \equiv \sum_i \frac{1}{(ER_i)^2}\left(\rho_{exp_i} - \alpha NL_{cal_i} - \beta L_{cal_i}\right)^2$$

where ER_i is the experimental error for measurement i

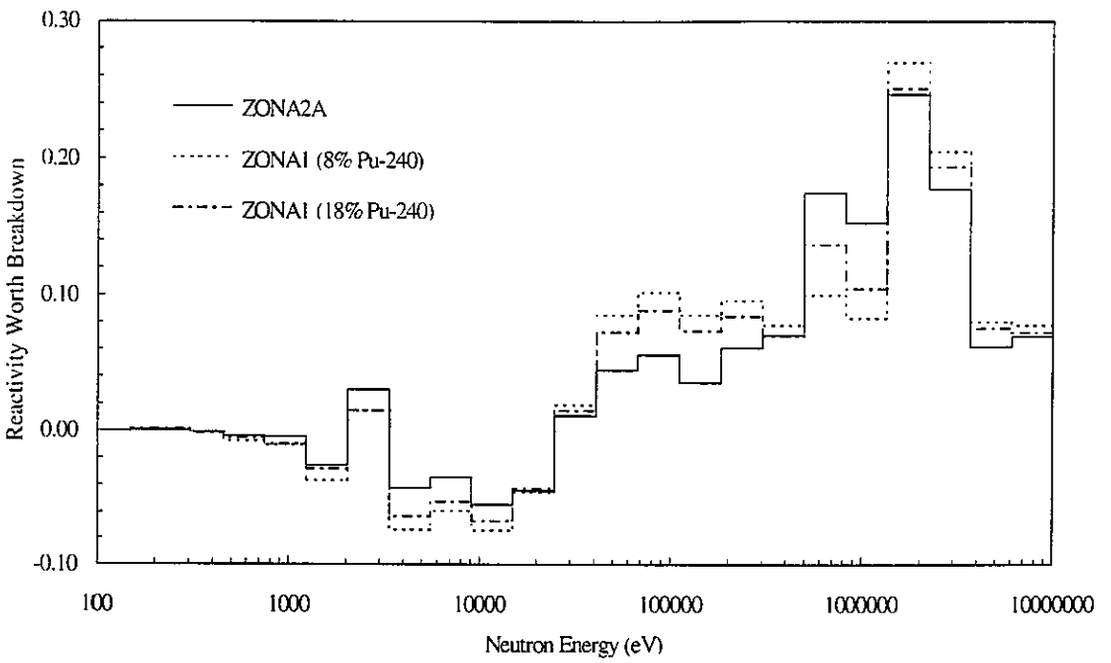
The resulting reconstructed C/E values for the two components of the sodium void reactivity worth are shown in the following table :

	Method	ZONA2A	ZONA1/R1 8 % Pu240	ZONA1/R1 18 % Pu240
Non-leakage term	Diffusion	0.92±0.05	1.08±0.03	1.02±0.05
	S4 P0	1.04±0.05	1.17±0.03	1.10±0.05
	S4 P1	1.00±0.05	1.14±0.05	1.07±0.05
Leakage term	Diffusion	0.91±0.05	0.94±0.05	0.89±0.05
	S4 P0	0.84±0.05	0.87±0.05	0.82±0.05
	S4 P1	0.84±0.05	0.87±0.05	0.83±0.05

It can be seen that for the CIRANO ZONA2A configuration the non-leakage term is well calculated when using transport theory, while for the ZONA1/R1 configuration it is overestimated. A breakdown of the reactivity worth by energy group (Figure 5) indicates that these discrepancies may be caused by the adjoint flux shape which is dependant on the ratio of the JEF2 absorption and fission cross sections for the heavy nuclides at ~ 1 MeV, an effect which is also supported by sensitivity analyses.

For all of the configurations studied it was found that the leakage term is underestimated by more than 10 %. Previous analyses have shown that this is caused by a strong streaming effect associated with the existence of two diagonal corridors in the MASURCA subassembly design. It is estimated[13] that this streaming effect increases the leakage term by approximately 10 %. This problem can be resolved by the implementation of already existing routines in the current calculation scheme. The transposition of the experimental results to large fast reactor designs is possible because of the limited streaming effect for the leakage component in fast reactor hexagonal sub assemblies[17].

FIGURE 5 : Energy Spectrum for the Non-Leakage Term



CONCLUSION

The adequacy of the JEF2/ECCO/ERANOS system for calculating sodium void reactivity worth in the ZONA2A core of the CIRANO experimental programme has been investigated. The results of this analysis have been compared with those obtained from a similar analysis performed for the ZONA1/R1 (8 % Pu240) and ZONA1/R1 (18 % Pu240) configurations, and a breakdown has been given for the non-leakage and leakage terms respectively.

It has been shown that for the CIRANO ZONA2A configuration the non-leakage term is well calculated when using transport theory, while for the ZONA1/R1 configurations it is overestimated. The leakage term is underestimated by more than 10 % in all of the configurations studied, caused by the existence of a strong anisotropy effect.

The resolution of these discrepancies in the current scheme requires :

- the adjustment[15] of the JEF2 cross section data evaluation based on measurements performed in experimental configurations with different plutonium enrichments.
- the development of an exact 2-dimensional capability within the ECCO cell code and the TGV nodal transport code for treating the anisotropy of the streaming.

REFERENCES

1. J.L. ROWLANDS, N. TUBBS
The Joint Evaluated File: a new nuclear data library for reactor calculations.
Santa Fe Symposium 1985

2. M.J. GRIMSTONE, J.D. TULLETT AND G. RIMPAULT
Accurate treatments of Fast Reactor Fuel Assembly Heterogeneity in the ECCO Cell Code. PHYSOR 90, Marseille, France, June (1990).

3. J.Y. DORIATH, C.W. Mc CALLIEN, E. KIEFHABER, U. WEHMANN, J.M. RIEUNIER
ERANOS 1 : the Advanced European System of Codes for Reactor Physics Calculation, International Conference on Mathematical Methods and Super Computing in Nuclear Applications, 19-23 April 1993, Kongresszentrum, Karlsruhe Germany.

4. P.J. FINCK, J.C. GARNIER and J.C. CABRILLAT
CAPRA Physics and the CIRANO Experimental Programme in Advanced Fuel Cycle, PSI Workshop, Villigen, Switzerland, September 18-19 (1995).

5. J. ROUAULT ET AL.
Physics of plutonium burning in Fast Reactors. Impact on Burner Cores Design.
Topical Meeting on Advances in Reactor Physics
KNOXVILLE - TENNESSEE, April 11-15, 1994.

6. F. LYON, M. MARTINI, G. RIMPAULT
Etude de l'effet de vidange sodium dans les milieux représentatifs des centrales à neutrons rapides de type classique ou hétérogène : Expériences faites au cours du programme PRE-RACINE sur MASURCA. Colloque sur la physique des réacteurs à neutrons rapides. Aix en Provence, France, AIEA-SM-244, 23 Septembre 1979.

7. C.J. DEAN, C.R. EATON, P. PEERANI, P. RIBON, G. RIMPAULT
Production of Fine Group Data for the ECCO Code. International Conference on the Physics of Reactors Operation Design and Computation PHYSOR 90, April 23-27, 1990.

8. R.E. MAC FARLANE, D.V. MUIR, R.M. BOICOURT
The NJOY Nuclear Data Processing System
LA - 9303 - M(ENDF - 324), 1982.

9. P. RIBON, J.M. MAILLARD
Probability Tables and Gauss Quadrature : Application to Neutron Cross-sections in the unresolved energy range. ANS Topical Meeting on Advances in Reactor Physics and Safety, Saratoga Springs, September 1986.

10. P. BENOIST
Streaming Effects and Collision Probabilities in Lattices,
Nucl. Sci. Eng., 34, 285, 1968.

11. G. RIMPAULT
Algorithmic Features of the ECCO Cell Code for Treating Heterogeneous Fast Reactor Assemblies. International Conference on Mathematics and Computation, Reactor Physics and Environmental Analysis, May 1995, Portland, Oregon.

12. P. BENOIST, J. MONDOT, I. PETROVIC
Calculational and experimental investigations of void effect. A new model for leakage treatment of heterogeneous assemblies. Nucl. Sci. and Eng., 118, 197 (1994)

13. C.J. GHO
Homogénéisation du Coefficient de Diffusion Influence de la Modélisation et du Laplacien pour les Réacteurs Rapides de Puissance et les Maquettes Expérimentales. PhD. Thesis of the Grenoble University (1984) France.

14. J.Y. DORIATH, F. MALVAGI, G. PALMIOTTI, J.M. RUGGIERI, C.B. CARRICO, E.E. LEWIS, G. GASTALDO
Variational Nodal Method (VNM) to solve 3-D transport equation. Application to EFR design. International Conference on Mathematical Methods and Super Computing in Nuclear Applications, 19-23 April 1993, Kongresszentrum, Karlsruhe Germany.

15. E. FORT, W ASSAL, G. RIMPAULT, J.L. ROWLANDS, P. SMITH, R. SOULE
Realisation and Performance of the Adjusted Nuclear Data Library ERALIB1 for Calculating Fast Reactor Neutronics. To be presented at this conference.

16. G. RIMPAULT, S. PELLONI, P. SMITH et al
Experimental Validation of Nuclear Data and Methods for Steel Reflected Pu Burning Fast Reactors. To be presented at this conference.

17. P. BENOIST, T. DURACZ
Diffusion Coefficients for Fast Reactor Hexagonal Assemblies.
Nucl. Sci. and Eng., 87, 72-102 (1984)