VENUS-7 Plutonium Recycling Benchmark, Results of AREVA NP

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Abstract

Solutions for the NEA VENUS-7 plutonium recycling benchmark are presented in this paper. Various few-group 3D transport calculations were performed with pin cell homogenized cross sections, mostly generated by CASMO-4 ("L-Lib" based on ENDF/B data). In addition, also 2D solutions with a finer energy group structure are presented. In general the calculated reactivity effects agree well with the measured ones. A comparison with other VENUS configurations indicates that the reactivity of the MOX pins with Inconel 800 cladding seems to be slightly undererestimated. The calculated fission rates in the VENUS-7/1 configurations show good agreement with the measured fission rate traverses. This is also confirmed by a VENUS-9/0 analysis where preliminary measured fission rate data were available also at the water reflector, displaying the strong peaking at this reflector boundary.
Keywords: VENUS-7 Benchmark, 3D transport solutions, reactivity effects, fission distributions

1. INTRODUCTION

Within the framework of the joint activities of the OECD/NEA Expert Group on Reactor-based Plutonium Disposition (TFRPD) and the Working Party on Scientific Issues in Reactor Systems (WPRS), various theoretical and experiment-based benchmarks have been organized. In 2005 the VENUS-7 configurations have been selected as an additional benchmark exercise [1]. The VENUS-7 core represents a valuable benchmark since it constitutes a full (though small) core with relatively high enriched fuel with different MOX fuel contents and with different Pu isotopic compositions. In total, three sub-configurations have been selected for the benchmark:

- VENUS-7/0 which was basically designed for a comparative study on reactivity of different types of MOX fuel with various ^{240}Pu and ^{241}Pu contents. The different types of MOX fuel are inserted in a central substitution zone.
- VENUS-7/1 where the reactivity of peripheral fuel rods was measured. To some extent also radial fission rate traverses have been measured.
- VENUS-7/3 which is a core with a reduced amount of MOX rods and with higher radial leakage

These core configurations are shown in Fig.1 (in quarter core configuration). The configurations mainly consist of 4/0 fuel (UO_2 with 4 w/o U-235 enrichment with stainless steel cladding) at the periphery, and 3/1 fuel (MOX with 3 w/o U-235 enrichment and 1 w/o Pu with high Pu quality (92 w/o Pu-239) with Inconel 800 cladding) in the core centre. In VENUS-7/0 a small region of 4x4 rod positions was foreseen as a substitution zone for various fissile materials to measure their reactivity effect: Besides the 4/0 and 3/1 fuel mentioned above, two further materials were evaluated with respect to their reactivity:

- A MOX fuel type "2/2.7" with 2 w/o U-235 and 2.7 w/o Pu (79 w/o Pu-239) and stainless steel cladding
- Another MOX 3/1 type ("0.38 swaged") with identical linear fuel density as the other 3/1 MOX fuel but with thicker Inconel 800 cladding.

2. LATTICE CALCULATIONS / CROSS SECTION DATA

In the present paper different solutions by AREVA NP for VENUS-7 are presented. Most solutions are based on cross sections generated with CASMO-4 [2]. This spectral code has been used both as a cell code (with the VENUS-7 core being simulated as a large 2D macrocell with uniform axial buckling) and as cross section library generator for various 3D transport codes. Using CASMO-4 for cross section generation is presently one of the standard procedures for AREVA NP core code systems such as CASCADE-3D (PWR) or COMPASS (BWR). However, two-group diffusion core calculations as implemented in the 3-D simulators of those code systems (e.g. MICROBURN-B2) are not accurate enough for the small VENUS-7 core. Therefore, for simulation of the VENUS-7 core, only transport solutions will be presented. The number of condensed energy groups is extended to 5 groups (two fast, one epithermal and two thermal groups). Furthermore only pin cell averaged cross sections are used in the 3-D calculations. The five group structure together with the assumed fission spectrum is given in Table 1.

Table 1: Energy Group Structure in 3D Core Calculations

group	upper boundary	Fission spectrum
1	10.00 MeV	0.785
2	821.0 keV	0.215
3	5.53 keV	0.0
4	4.00 eV	0.0
5	0.625 eV	0.0

As an alternative, cross sections in 44 groups have been generated using SCALE5 [3] based on ENDF/B6. The following table shows the CASMO-4/SCALE5 k-infinity values for the specified fuel zones and the corresponding values of condensed homogeneous cross sections:

Table 2: k-infinity values of fissile materials

	CASMO-4	CASMO-4 condensed	SCALE5
	k-inf (70 group)	k-inf (5 group)	k-inf (44 group)
UO_2 4/0	1.34887	1.347757	1.34587
MOX 3/1	1.31583	1.31462	1.31163
MOX 3/1 0.38 swaged	1.27997	1.279207	1.27753
MOX 2.0/2.7	1.27543	1.2745	1.27256

The homogenized cross sections for the non-fissile materials (water reflector, axial reflectors) are derived from appropriate simple CASMO-4 cell models with fuel environment. The CASMO-4 k-infinity values are calculated with the "L-lib", based on ENDF/B data in 70 energy groups. For the pin cell row directly adjacent to the water reflector, separate cross section sets have been generated (CASMO-4 five group condensation) in order to take into account the stronger flux thermalization in the vicinity of the water reflector. The results show that the k-infinity values calculated by SCALE5 (44 group library) are systematically around 300 pcm lower than those of CASMO-4.

3. REACTIVITY EFFECTS IN VENUS-7

In the core calculations CASMO-4 was used to generate a deterministic 2D transport solution (quarter core) with a rather fine energy resolution (70 groups). In addition, the VENUS-7 configurations were analyzed with three 3-D transport codes:

1) The broad energy group Monte-Carlo transport code MOCA [4]. Monte Carlo codes generate a reliable “best estimate” 3-D k_{eff} level for a given cross section data set. On the other hand, these calculations are not ideal for calculating small reactivity effects of the order of 100 pcm (as in VENUS-7) if the statistical error (1σ) of the Monte Carlo method is of the order of 10 pcm (which is what can be achieved with reasonably low running times using Monte Carlo methods).
2) The 3D discrete ordinate code TORT developed by ORNL [5]. Here an important aspect is to choose a reasonable mesh sizing. It turned out that satisfactory solutions (close the Monte Carlo eigenvalues) could be achieved by S_8 quadrature order, mesh sizing of 1 cm axially and 2x2 meshes per pin cell (1.3 cm being the pin pitch). In addition to the 3D solution based on CASMO-4 cross sections also, the corresponding 2D code DORT was used to calculate a solution with S_8P_3 in 44 energy groups (cross sections being generated with SCALE5).
3) The fast nodal transport code VARIANT (DIF3D-8.0) developed by ANL [6]. Here the option P_3 with “simplified spherical harmonics” is chosen which represents a very good compromise between low running times and high accuracy (actually, with this option VARIANT is very close to the full transport solution and displays an excellent convergence behavior). One node per pin cell radially and eight nodes axially in the core were chosen for the core model in VARIANT. Only cross sections in 5 groups from CASMO-4 were used in VARIANT.

Both DORT/TORT and VARIANT codes (which are not standard AREVA NP core design programs) are available via the NEA data base.

The following table gives an overview of all calculated eigenvalues for the VENUS-7 configurations (with the deterministic “fine-meshed” TORT results taken as “reference”):

Table 3: k_{eff} values of various core configurations

Case / k_{eff}	CASMO-4 (2D, 70 Gr)	TORT (3D)	VARIANT (3D) P_3	TORT-VARIANT	MOCA (MC Ref.)	TORT - MOCA	TORT – CASMO-4
VENUS-7/0 (M3/1)	0.99308	0.995364	0.9948654	50 pcm	0.995627	-26 pcm	228 pcm
VENUS-7/0 U4/0	0.99419	0.996571	0.9960635	51 pcm	0.996694	-12 pcm	238 pcm
VENUS-7/0 M2/2.7	0.99225	0.994608	0.9941098	50 pcm	0.994864	-26 pcm	236 pcm
VENUS-7/0 M3/1 0.38swaged	0.99225	0.994602	0.9940923	51 pcm	0.994801	-20 pcm	235 pcm
VENUS-7/1	0.99416	0.996396	0.995922	47 pcm	0.996784	-39 pcm	224 pcm
VENUS-7/1 – 4 rods	0.99364	0.995895	0.995415	48 pcm	0.996035	-14 pcm	226 pcm
VENUS-7/3	0.9920	0.995437	0.994773	66 pcm	0.995527	-9 pcm	344 pcm
Comparison with other VENUS configurations:							
VENUS 1			0.997720		0.99858		
VENUS 2			1.000252		1.00110		
VENUS 9/0			0.999089		0.99983		
VENUS 17/0			0.999114		1.00004		

Both TORT and VARIANT yield k_{eff} values which are very close to the Monte Carlo values (lower than 100 pcm difference). The general k_{eff} level is around 0.995 which is slightly lower (about 300-500 pcm) than for other VENUS configurations with identical methods. The 2D k_{eff} values – calculated with a uniform axial buckling of 0.0024 cm^{-2} - are slightly lower than the 3D ones. The eigenvalues of the 2D solution with DORT/SCALE are not included here. They are around 300 pcm lower than the CASMO-4 values.

The calculated reactivity effects of the various fuel types in the substitution zone (VENUS-7/0) which can be derived from the eigenvalues above are compiled in the following table:

Table 4: Reactivity effects [pcm] in substitution experiment (VENUS-7/0)

fuel type in substitution	**CASMO-4**	**TORT**	**VARIANT**	**MOCA (1σ)**	***measured***
U 4/0	**111**	**121**	**120**	**107 ± 15**	***94 ± 4***
MOX 2/2.7	**-83**	**-76**	**-76**	**-76 ± 15**	***-92 ± 4***
MOX 3/1 0.38swaged	**-83**	**-76**	**-77**	**-83 ± 15**	***-58 ± 4***
U 4/0 - MOX 2/2.7	**194**	**196**	**195**	**183 ± 15**	***186 ± 6***

In general these reactivity effects agree quite well with the measured ones /1/. It is noteworthy that the reactivity of the MOX rods of the type 3/1 (i.e. 3 w/o U-235, 1 w/o Pu-fiss) appears to be slightly underestimated compared to the 4/0 and 2/2.7 fuel rods, which is also indicated by the general k_{eff} level of other VENUS configurations (with k_{eff} being closer to one, see table 3). One reason for this behaviour may be the different cladding material of the 3/1 rods (Inconel 800) compared to stainless steel in the other fuel rods. Apart from VENUS 9, this cladding material is not present in the other VENUS configurations. This could indicate that the strongly absorbing isotopes in Inconel 800 (Co and Mn) have lower concentrations than specified in the benchmark.

In VENUS-7/1 the reactivity worth of one peripheral rod (13 pcm with the deterministic codes) is somewhat underestimated, compared to the experimental value of 16 pcm. This may be due to the location of the peripheral rod specified in the benchmark (the exact position in the experiment not being exactly known). When choosing the most reactive position of the peripheral rods instead, the calculated and measured reactivity worth values agree well (16.6 pcm in CASMO-4).

4. FISSION RATE DISTRIBUTIONS

In the VENUS-7/1 experiment fission rate distributions were measured on a small scale (rates along central symmetry lines). Only data for the central 3/1 zone are available, therefore only for this central zone the measured and calculated fission rate traverses can be compared. As Fig. 2 indicates the calculated fission rate distribution agree very well with the measured values (1σ deviation <1% for all solutions, from fuel rod no. 1 to 10 in Fig.2). Both the radial buckling and the behavior at the boundary to the U 4/0 zone is well described in all solutions. Here the TORT solution has been omitted, since it is almost identical with VARIANT results. Another important aspect, which was not covered by the experimental data in VENUS-7, is the strong peaking of the fission distribution in the peripheral row at the water reflector (rod no. 17 in Fig.2). This strong peaking is observed in a similar manner in all solutions. In fact the solutions for the full reactor core differ only very little: the 1σ deviation is less than 1% between CASMO-4 and DORT/SCALE for all rods in the core and around 1.5% between VARIANT and CASMO-4. A comparison with an MCNP solution of GRS based on JEF2.2 [7] has shown that the MCNP fission rate distribution is also very close to CASMO-4.
As the power peaking at the outer water reflector represents a great challenge for neutronic codes, also data from the VENUS 9/0 experiment are used in the present paper to evaluate this effect in more detail. Only preliminary data for VENUS 9 are available to the authors and these are not yet fully authorized by NEA. The VENUS 9/0 configuration is a very simple rectangular core with essentially the same two materials (3/1 and 4/0) as in VENUS-7. The 2D geometry (rod distribution) is shown in Fig. 3 where also the measured fission rate traverse is indicated. In VENUS 9/0 this includes both 4/0 UO_2 fuel and 3/1 MOX fuel, covering both the internal boundary between these two fuel types and the water reflection at the outer boundaries.
The results for the fission rate distributions are shown in Fig.4 (UO_2 zone) and Fig.5 (MOX 3/1 zone). The calculations were performed using homogenized pin cell cross sections, in 3D with VARIANT (5 group cross sections generated by CASMO-4) and in 2D with DORT (44 group cross sections generated by SCALE5). The agreement between the measured and calculated fission rate distributions is generally excellent (1σ deviations between 1.0 and 1.4% for the individual zones, being clearly not larger than the experimental uncertainty). Both the behavior at the boundary between the two fuel zones and at the periphery with water reflection is calculated with a good accuracy.

CONCLUSIONS

The VENUS-7 experiments which have been made available to the nuclear community by the NEA constitute a very valuable experimental data set for validation of nuclear codes. The calculations presented here by AREVA NP show that a few-group approach with pin cell homogenized cross sections gives good results for 3D transport calculations regarding both k_{eff}

and pin power distributions, even for a small high leakage core such as VENUS-7 with partially loaded MOX fuel. Two dimensional methods with finer energy resolution (CASMO-4/DORT) appear to yield a slightly better fission rate distribution but are less accurate regarding core k_{eff}. The strong peaking of the fuel rod fission rate at the core periphery with water reflection is calculated accurately with the present methods as the comparison with the preliminary VENUS 9 measurement data shows.
The VENUS-7 experiments will be part of the NEA IRPhBEP project [8]. They will also provide a valuable test for future AREVA NP codes such as APOLLO-2A.

REFERENCES:

1. Benchmark on the VENUS Plutonium Recycling Experiments – Configuration 7
NEA/NSC/DOC(2005) 11, http://www.nea.fr/
2. M. Edenius, B. H. Forssen and C. Gragg,
"The Physics Model of CASMO-4", Proc. Adv. in Math., Comp. & Reactor Physics, Vol. 10-1, Pittsburgh (1991)
3. SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations,
ORNL/TM-2005/39, Version 5, Vols. I-III, April 2005.
Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-725.
4. J. Lieberoth
"A Monte Carlo technique to solve the static eigenvalue problem of the Boltzman transport equation"; Nukleonik 11, 213-219 (1968)
5 Rhoades, Childs
"TORT-DORT Two- and Three-Dimensional Discrete Ordinates Transport Version 2.7.3", RSIC CODE PACKAGE CCC-543
ORNL, Oak Ridge, Tennessee, 1993
6. G. Palmiotti, C. B. Carrico, and E. E. Lewis,
"Variational Nodal Methods with Anisotropic Scattering"
Nuclear Science and Engineering, 115, 233-243, 11/93.
7. W. Zwermann (GRS Garching), private communication
8. International Reactor Physics Benchmark Experiment Evaluation Project (IRPhBEP)
NEA/Idaho National Laboratory
current status: NEA/NSC/DOC(2006)1

Figure 1: Various VENUS-7 configurations
(2D quarter core, center bottom right, all configurations with full water reflection)

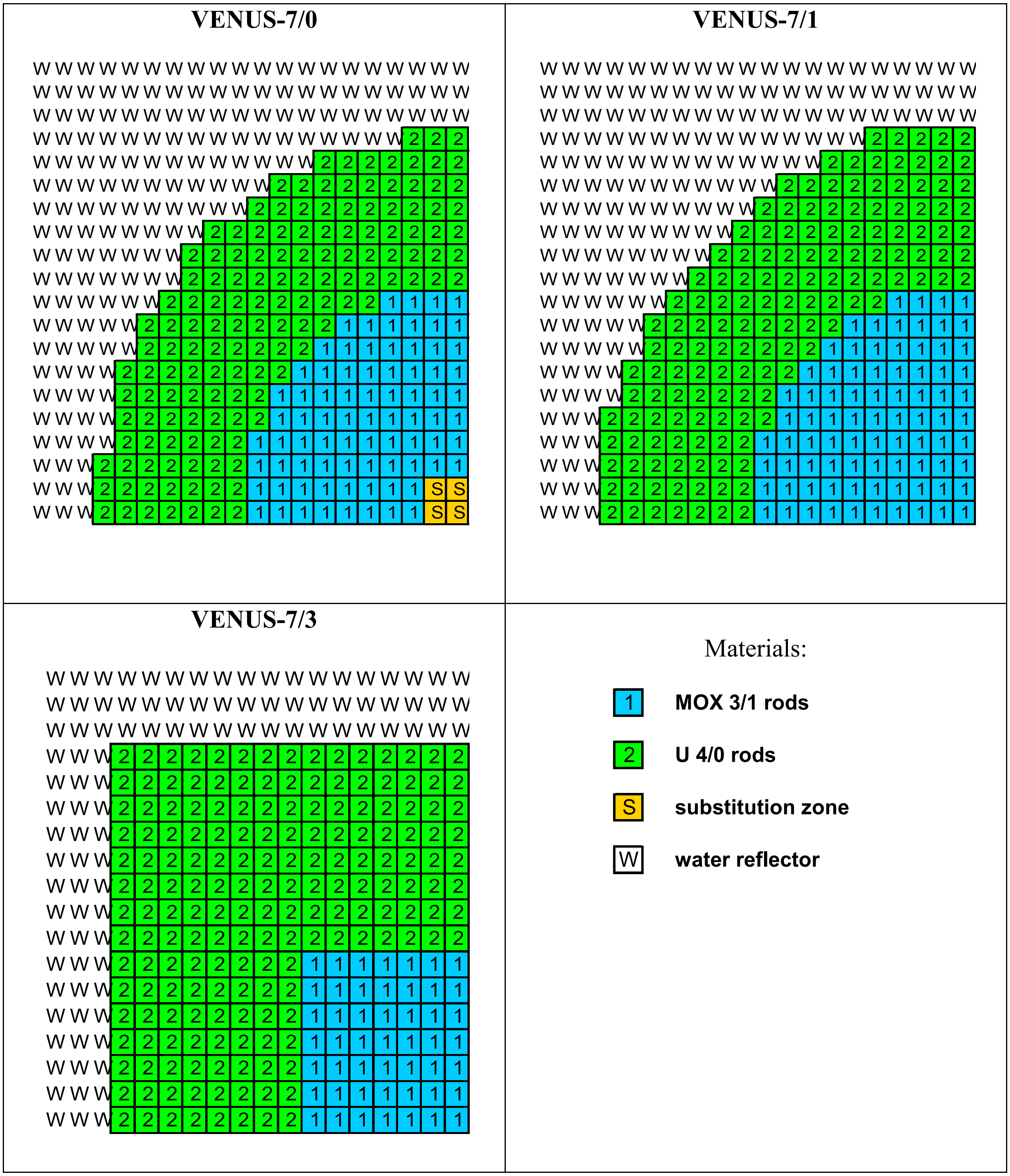

Figure 2: VENUS-7/1 fission rate distribution
(rod no.1 = core center, rod no. 17 = peripheral rod with water reflection)

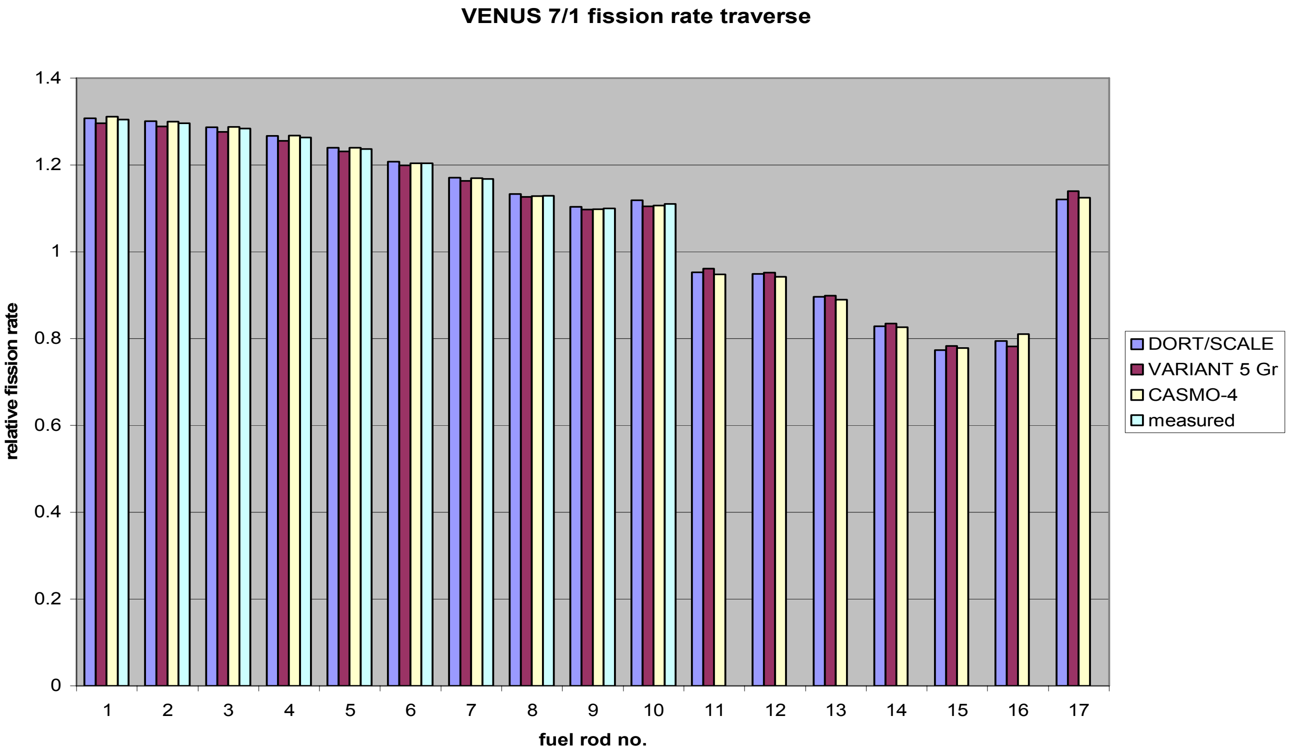

Figure 3 VENUS 9/0 core layout, fuel rod distribution
(1 = MOX 3/1 fuel, 2 = U 4/0 fuel, boxes with bold borderline: measured fission rates)

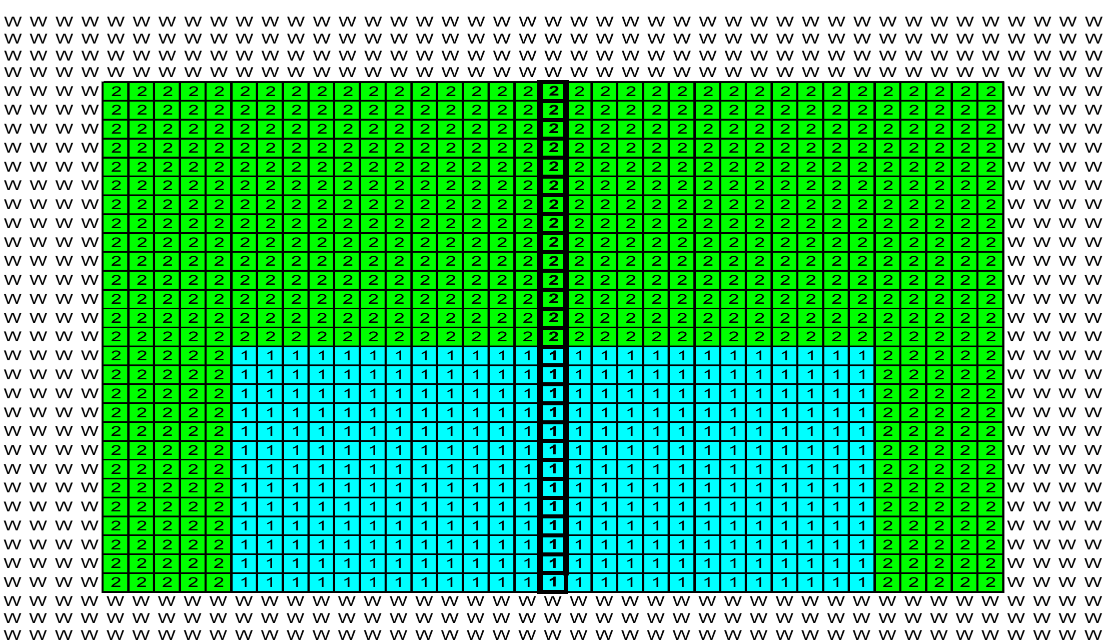

Figure. 4 VENUS 9/0 fission rate distributions in UO_2 4/0 zone
(rod no.1: core periphery, boundary to water refl., rod no. 14: core centre,boundary to MOX 3/1 fuel)

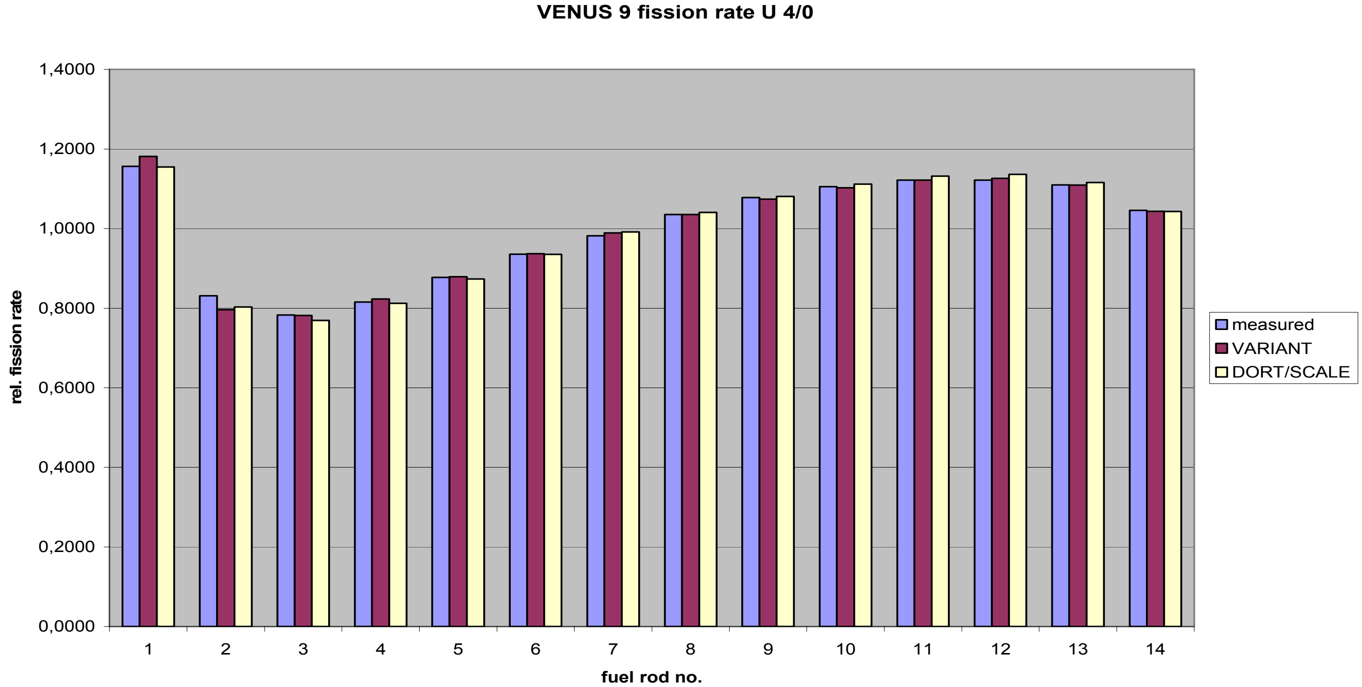

Figure 5 VENUS 9/0 fission rate distributions in MOX 3/1 zone
(rod no.1: centre, boundary to U 4/0 zone, rod no. 13: core periphery, boundary to water refl.)