Benchmark Calculation on the VENUS-7 MOX Fuel Experiments

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1. Introduction

In the late 1960's and the early 1970's, SCK•CEN (Mol, Belgium) carried out a series of experiments entitled "Plutonium Recycling Physics Project" using the VENUS (Vulcain Experimental Nuclear Study) facility. More than 100 different core configurations have been studied. Recognizing a further need for validation of computing methods and nuclear data for MOX-fuelled systems, SCK•CEN has released all sets of these experimental results to the OECD/NEA for the international community. After having examined, the OECD/NEA expert groups have selected three most interesting configurations for international benchmark exercises. A series of the benchmarks based on these configurations and their experimental results will be organized by the OECD/NEA. One of these experimental data is the configuration VENUS-7 for different types of MOX fuel with different Pu contents. As a part of this benchmark program, 3-dimensional benchmark models have been established based on the problem specification for the configuration VENUS-7 by the Mote Carlo code MCNP4C [1] and the calculation results have been compared with some available VENUS-7 experimental data.

2. Method and Results

The VENUS-7 configuration was designed for a comparative study on different types of MOX fuel with different ^{240}Pu and ^{241}Pu contents. In total, four sub-configurations (7, 7/1, 7/2, and 7/3) were studied in which different nuclear parameters were measured. More detailed description on the benchmark experiments and requested calculations can be found in Reference 2.

On request by the OECD/NEA, nuclear parameters for the 3 sub-configurations (7, 7/1, and 7/3) have been evaluated by the MCNP4C code. A summary of the configuration descriptions, corresponding requested calculations, and their results are presented in the following sections.

2.1 Problem Specification

Configuration 7: comprises a cylindrical central zone with 332 rods containing heterogeneous vibrated 3/1 (3% ^{235}U/ 1% Pu) fuel, in which the 16 central rods were substituted by 4/0 UO_2 (4% ^{235}U/ 0% Pu) rods, by 3/1 type fuel from different fabrication processes and by 2/2.7 type MOX (2% ^{235}U/ 2.7% Pu) fuel. The peripheral area of the configuration comprised an annular region containing type 4/0 UO_2 rods, except in peripheral regions, where various rods were loaded in accordance with fuel availabilities. The surrounding reflector as well as the moderator consists of pure water at ambient temperature.

Configuration 7/1: comprises a cylindrical central area with heterogeneous vibrated 3/1 fuel in 332 rods and an annular zone of 580 fuel rods with 4/0 fuel rods.

Configuration 7/3: comprises a central square of 196 rods of type 3/1 vibrated heterogeneously, which is surrounded by a concentric square region with 704 rods of type 4/0.

2.2 Requested Calculation

For each sub-configurations, following benchmark calculation results were requested by the OECD/NEA.

Cell calculation: infinite multiplication factor (k_∞) for unit cell containing different types of fuel rod: 4/0 UO_2, 3/1 MOX (0.38 swaged), 3/1 MOX, and 2/2.7 MOX fuels. (Table 1)

Configuration 7: effective multiplication factor (k_{eff}), kinetic parameters (delayed neutron fraction β and neutron generation time Λ), and variation of k_{eff} due to the substitution of 16 central rods by 4/0 UO_2 rods, by 3/1 type fuel from different fabrication processes and by 2/2.7 type MOX fuel. (Tables 2 and 3)

Configuration 7/1: k_{eff} and rod worth of outermost 4 fuel rods. (Table 4)

Configuration 7/3: k_{eff}. (Table 5)

2.3 MCNP Modeling and Calculation

MCNP4C code was used to perform benchmark calculations for various VENUS-7 core configurations which utilized different types of MOX fuels for the experiment.

For general purpose of criticality calculations, continuous-energy cross-sections based on the ENDF/B-VI were used. To obtain neutron generation time for the configuration 7, a fixed source adjoint transport calculation was additionally performed using multi-group cross-sections based on the ENDF/B-V with 30-group neutron energy structure [3].

To obtain infinite multiplication factor for unit cell MCNP model, reflective boundary conditions for 4 sides with infinite z-axis were utilized. For the VENUS-7 configurations, full 3-D core was explicitly modeled in three-dimensional geometry. In order to facilitate the explicit modeling of fuel rods in the core, a repeated structure option of MCNP was used. Radial boundary of MCNP model was chosen for the inner boundary (60cm) of reactor support zone and axially the top and bottom reflector zones (50 cm < z < 168 cm) were included in the model.

The number of histories for criticality calculations for infinite cells and kinetic parameters was 1.0E7 (20,000 neutrons/cycle and 500 cycles after 50 inactive cycles). They were increased to 1.0E8 (200,000 neutrons/cycle and 500 cycles after 50 inactive cycles) for the various VENUS-7 configuration models to take into account of the model size. As for the fixed source adjoint calculation to obtain neutron generation time, the number of histories of 5.0E7 was used for the MCNP run.

2.4 Calculation Results

The calculational results are listed in the Tables 1 through 5 and the available experimental data are also listed and compared in the Tables.

Table 1. Unit cell: k_∞ values

$4/0\ UO_2$	3/1 MOX 0.38 swaged	3/1 MOX	2/2.7 MOX
1.35097 (±0.00018)*	1.27818 (±0.00019)	1.31109 (±0.00019)	1.27414 (±0.00019)

* Estimated standard deviation (1σ)

Table 2. Configuration 7: k_{eff} and kinetic parameters

Parameter	Calculated	Experimented	(C-E)/E
k_{eff}	0.98742±0.00008	-	-
Λ	2.1537E-5 sec	-	-
β	657.344 pcm	-	-
Λ/β	3.276E-3 sec	3.20E-3 sec	2.4%

Table 3. Configuration 7: Substitution test results in the core center

Parameter	Δk_{eff}* (%)		
	C**	E***	(C-E)/E
$4/0\ UO_2$	+0.144	+0.094	53.2
3/1 MOX, 0.38 swaged	-0.055	-0.058	-5.2
3/1 MOX, 0.27 swaged	+0.022	-	-
2/2.7 MOX	-0.055	-0.092	-40.2

*$\Delta k_{eff} = k_{eff}$(substituted)- k_{eff}(standard conf. 7)
**Calculated values, all calculated k_{eff} have standard deviations of 0.00008
***Experimented value

Table 4. Configuration 7/1: k_{eff} and peripheral rod worth

Parameter	Calculated	Experimented	(C-E)/E
k_{eff}	0.98855±0.00007	1.0009	-1.2%
Δk_{eff}/pin*	12.75 pcm	16 pcm	-20.3%

*Difference of k_{eff} values before and after removing 4 peripheral rods divided by 4

Table 5. Configuration 7/2: k_{eff} value

Parameter	Calculated	Experimented	(C-E)/E
k_{eff}	0.98816±0.00007	0.9985	-1.0%

Comparing several parameters obtained from VENUS-7 experiments and MCNP benchmark calculations, there are some discrepancies for Δk_{eff} values due to the slight change in the core configuration as for the substitution test of configuration 7 and rod worth test of configuration 7/1. Considering the reasons for the discrepancies, it can be inferred from the fact that the system parameters (k, k′) have relatively large statistical errors compared to the small variation (Δk=k-k′) of the system configurations, which leads to the high relative differences (C/E-1) between calculation and experiment results. The estimated standard deviations (1σ) of the k_{eff} amount to 6%~16% of Δk_{eff} as found in Tables 3 and 4.

3. Conclusion

Benchmark calculations for the VENUS-7 core configurations requested by the OECE/NEA relating to plutonium recycling physics experiments have been performed by MCNP4C code in MS-DOS computing environment.

The VENUS-7 MOX fuel benchmark is a “blind” benchmark but some of the experimental results were listed in the OECD/NEA internal document [2]. Owing to the information about experimental data, the performed benchmark calculations can be compared with experimental results.

The calculated system parameters (k) and kinetic parameters (Λ/β) show good agreement with those obtained from experiment. But as for the calculations of searching small variations (Δk) due to the slight change in the core configuration, it is found difficult to obtain satisfactory results by just differencing the MCNP results before and after the configuration change due to the statistical errors inherent to the Monte Carlo method.

REFERENCES

[1] J. F. Briesmeister (ed.), “MCNPTM-A General Monte Carlo N-Particle Transport Code Version 4C,” LA-13709-M, Los Alamos National Laboratory, Mar. 2000.
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