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Simulating the Behaviour of the Fast Reactor JOYO (Draft)

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ABSTRACT

Motivated by the development of fast reactors the behaviour of the Japanese experimental fast reactor JOYO is simulated with two Monte Carlo codes: Monte Carlo N-Particle (MCNP) and Probabilistic Scattering Game (PSG). The simulations are based on the benchmark study "Japan's Experimental Fast Reactor JOYO MK-I core: Sodium-Cooled Uranium-Plutonium Mixed Oxide Fueled
Fast Core Surrounded by UO2 Blanket". The study is focused on the criticality of the reactor, control rod worth, sodium void reactivity and isothermal temperature coefficient of the reactor. These features are calculated by applying both homogeneous and heterogeneous reactor core models that are built according to the benchmark instructions. The results of the two models obtained by the two codes are compared with each other and especially with the experimental results presented in the benchmark.

1 INTRODUCTION

If any uranium-based nuclear power is going to be deployed also in the remote future, the introduction of reactors with fast neutron spectrum will be necessary. Therefore, almost all major industrial countries have contributed to the fast breeder reactor (FBR) development over the decades since the beginning of the nuclear era. The high costs, technological difficulties and public resistance accompanied by nuclear proliferation concerns have caused some of the countries to bury their FBR programs. To some extent, the target of the efforts has shifted from the plan of producing energy to transmuting the spent fuel, also known as nuclear waste, to less harmful form. However, the fast breeder reactors would provide a method to generate large amounts of electricity, if the technology was developed to an adequate level to tackle the problems previously listed.

One of the nations still aiming at producing energy by fast reactors is Japan, whose FBR program has been motivated by the scarcity of domestic energy resources. The experimental fast breeder reactor JOYO has operated as a part of the Japanese fast reactor development program since 1977. The experiments referred to in the present study were completed in the beginning of the operation, originally without any significant intentions to create an ideal benchmark. However, the results and measurement conditions of the series of experiments have been later determined to be sufficient to allow the use of them as a benchmark. For the present study it signifies that the results can be used for validating computer simulation codes by simulating the experiments and comparing the experimental and computational results. The codes employed here are the MCNP (version 4C) and PSG. It is observed that in some cases the characteristics of the reactor may be estimated quite accurately, but also that the suitability of the Monte Carlo method encounters certain limits.

2 COMPUTATIONAL METHODS

Traditionally the life of the neutrons in reactor has been calculated by employing deterministic transport codes only. Following the continuous improvements of computational power and algorithms the use of Monte Carlo simulation codes has become increasingly suitable option in reactor physics modelling, although the dominance of deterministic codes probably continues a long time. The

deterministic codes are based on the procedure to solve valid neutron transport equations by using discretised values of the variables that are usually the space and momentum of the particles. All particles between two discretisation nodes thus have the same characteristics. In Monte Carlo method, for its part, every particle is treated as an individual, whose movements, reactions and lifetimes are determined by sampling such that the laws of physics and experimentally obtained statistical distributions are obeyed.

Something about the codes and their differences will be written here

3 THE JOYO REACTOR

The sodium-cooled experimental fast reactor JOYO achieved its first criticality in 1977. Over the first operational years the core model used was called MK-I. The power was produced in the core fuel region that was surrounded by blanket fuel region. The core was fuelled by mixed-oxide (MOX) fuel that consisted of 23% enriched uranium and plutonium that accounted for 17.7% of the weight of all metallic material. The plutonium content for its part included 80.4% fissile plutonium. The material in the blanket region consisted of depleted uranium including 0.2% fissile ^{235}U. The blanket region was also known as the breeding zone as the breeding of fissionable ^{238}U to fissile ^{239}Pu occurred mostly there. The leakage of the neutrons that were not captured in the blanket fuel was tried to stop by reflectors, both removable and fixed. At first the maximum thermal power of the reactor was 50 MW, but it was soon upgraded to 75 MWt. Since 2003 the plant has been operating with the maximum thermal power of 140 MW subsequent to the upgrade to MK-III core.

The experiments of the study were mostly carried out for 64- and 70-fuel-subassembly cores, but also an assembly with 65 fuel subassemblies was employed when measuring the isothermal temperature coefficient. The Figure 1 presents a core pattern of a 70-fuel-subassembly core. In various experiments the exact locations of some core and blanket fuel subassemblies at the border of the two regions were slightly altered, but no major modifications were conducted.

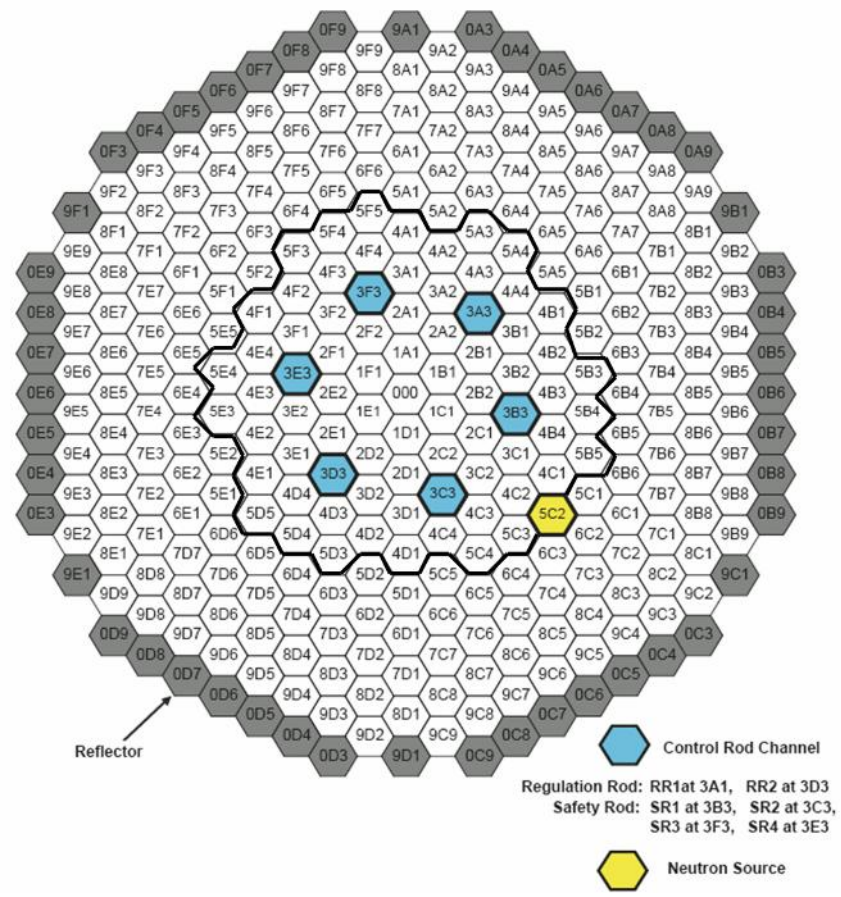

Figure 1: The horizontal cross-section of the 70-fuel-subassembly of Joyo. The core and blanket fuel regions are separated by the bold black line. This particular assembly was employed in sodium void reactivity measurements. In some other measurements the assembly was slightly different. [1]

4 SIMULATIONS (PRELIMINARY RESULTS)

4.1 Homogeneous model

In homogeneous model the material included in a subassembly was supposed to be evenly mixed in the subassembly level. The MCNP simulations were completed by 500 active calculation cycles preceded by 50 inactive ones. The initial neutron population of 10 000 was used in all simulations.

So far only the simulations of homogeneous model by MCNP have been completed.

4.1.1 Criticality

The first criticality of Joyo was achieved by adding core fuel subassemblies one-by-one with all control and safety rods, except for RR2, fully withdrawn until the criticality was reached. The milestone was broken with 64 fuel subassemblies at the temperature of 204.7°C. For the simplicity of calculations they are not completed at this temperature but at 250°C and with all rods completely withdrawn. The increase in temperature decreases and removal of RR2 increases the multiplication factor k, which must be taken into account. This is done by applying the isothermal temperature coefficient and the reactivity curve of RR2 the measurements of which will be described later. Also the

effects of the homogeneous model approximation must be considered when comparing the experimental results to the calculatory ones. With these corrections the comparable experimental value and thus the desired result of simulation becomes k = 0.9921. All experimental results in the following tables have been corrected to be comparable to the simulations, unless otherwise mentioned.

In the 70-fuel-subassembly core the criticality was achieved at the constant temperature of 200°C by varying the position of regulation rods. All safety rods were fully withdrawn and the criticality was achieved with regulation rods at about their half insertion. When adjusted the measured value to be comparable with the simulation, k = 0.9897 is obtained. The experimental and calculational figures are summarized in Table 1. The third row of the table presents the simulation of the core with fully withdrawn control rods. The obtained value of *k* will be used as the reference *k* in later simulations.

Table 1: The criticality simulations. In RR positions the figure 70 cm stands for full withdrawal from the core.

Criticality simulations				Benchmark	
Core	RR1 (cm)	RR2	k	Measured k	(C-E)/E (%)
64	70	70	0.99223 ± 0.00026	0.9921 ± 0.0069	0.013
70	35	31.7	0.98618 ± 0.00026	0.9897 ± 0.0073	-0.356
70	70	70	1.01238 ± 0.00027		

4.1.2 Rod worth

The experimental measurement of control rod worths was carried out with period method for RR1 and replacement method for other regulation and safety rods. The period method utilizes the reactor period through the so called *inhour-equation* that is not discussed more thoroughly here. In the replacement method a rod with known reactivity is inserted to compensate for the withdrawal of the rod with unknown reactivity, after which the reactivity difference is measured. The computer simulation was completed in a more straightforward way obeying the benchmark instructions such that the reactivity worth for each rod was simulated by having the rod fully inserted and the resultant multiplication factor was compared to that with all rods fully withdrawn. This kind of method would not be possible experimentally, since the excess reactivity would lead into a runaway behaviour of the reactor. The reactivity measurements considered here were performed in the 70-fuel-subassembly core.

More quantitatively, the rod worth in cents is given by

$$\mathrm{CRW} = \frac{k_1 - k_2}{k_1 k_2} \cdot \frac{1}{\beta_{\mathrm{eff}}} \cdot 100 \,, \tag{1}$$

in which k_1 denotes the *k* of the core with fully withdrawn control rods, k_2 the *k* of the core with the specific control rod inserted and β_{eff} the fraction of delayed neutrons. For 70-fuel-subassembly the value β_{eff} = 0.0052151 was used in all measurements and simulations. Both of the simulated values of k include some statistical error whose quantity is the standard deviation (stdev). So the total error of the absolute CRW can be calculated by using the standard deviation Δk_i as follows:

$$\Delta(\mathrm{CRW}) = \frac{\Delta k_1}{k_1} + \frac{\Delta k_2}{k_2} \,. \tag{2}$$

Table 2: The results of rod worth simulations compared to the respective measurements

Rod worth simulations (rod worth in cents)				Benchmark	
Rod	k	Stdev	Rod worth	Measured RW	(C-E)/E (%)
RR1	0.98987	0.00026	430.155 ± 10.151	381.8 ± 35.5	12.67
RR2	0.99025	0.00026	422.721 ± 10.149	379.4 ± 39.5	11.42
SR1	0.9891	0.00026	445.235 ± 10.155	399.4 ± 44.7	11.48
SR2	0.98867	0.00027	453.667 ± 10.351	403 ± 41.1	12.57
SR3	0.98978	0.00026	431.916 ± 10.151	390.2 ± 3.7	10.69
SR4	0.98903	0.00026	446.607 ± 10.155	395.2 ± 39.9	13.01
β_{eff} = 0.0052151					

4.1.3 Sodium void reactivity

In the real life the sodium void reactivity (SVR) was measured by replacing the sodium content of one of the fuel subassemblies by helium and comparing the resultant reactivity to the one with ordinary fuel load. The reactivity difference was obtained from the rod worths of critical core. It was learned in section 2.5 that the effect of the voided volume depends on its location and to find out the dependence the location of the voided subassembly was varied. The position number 000 in Table 3 means the subassembly in the middle of the core, whereas 6F1 refers to the outermost voided subassembly that is located in the blanket fuel region. The other numbers refer to the positions between the two, see Figure 1. In simulations the regulation rods were kept at 34 cm from full insertion and the safety rods were fully withdrawn. The reactivity differences were obtained from the difference of the multiplication factor between the core including a voided subassembly and the reference core. In mathematical terms this can be expressed

$$\mathrm{SVR} = \frac{k_1 - k_2}{k_1 k_2} \cdot \frac{1}{\beta_{\text{eff}}} \cdot 100 \,, \tag{3}$$

where k_1 represents the multiplication factor of the reference core, k_2 the k of the core with sodium removed from the specified subassembly and β_{eff} is the fraction of delayed neutrons. As in case of the control rod worth simulations the error of absolute sodium void reactivity can be calculated by using the standard deviations of the simulation:

$$\Delta(\mathrm{SVR}) = \frac{\Delta k_1}{k_1} + \frac{\Delta k_2}{k_2} \,. \tag{4}$$

The SVR turned out to be so small that the error margins of both the simulation and experimental measurements are quite large compared to the reactivity values.

Table 3: The simulations and measurements of sodium void reactivity. It should be noted that the experimental value at the location 5F1 is not corrected to the homogeneous model but it is the direct result of the measurement.

Sodium void reactivity			Benchmark	
Void position	k	Simulation	Measured SVR	(C-E)/E (%)
k-reference	0.98685			
000	0.98644	-8.08 ± 10.30	-7.38 ± 2.51	9.62
1F1	0.98615	-13.79 ± 10.30	-8.12 ± 2.43	69.83
2F1	0.98594	-17.93 ± 10.11	-8.14 ± 2.39	120.27
3F1	0.98678	-1.38 ± 10.49	-6.13 ± 1.98	-77.49
4F1	0.9862	-12.81 ± 10.30	-5.95 ± 2.44	115.29

5F1	0.98674	-2.17 ± 10.30	-2.46	-11.79
6F1	0.98724	7.68 ± 10.10	0.56 ± 1.39	1271.43
β_{eff} = 0.0052151				

4.1.4 Isothermal temperature coefficient

The isothermal temperature coefficient (ITC) was measured at low power by changing the reactor temperature in between 170 and 250°C. The coefficient was measured for cores with both 65 and 70 fuel subassemblies, but the simulation was calculated only for the 65-core. Again the small value of the coefficient caused an accuracy problem for the simulation, since the error margins allow a very large range for the resultant coefficient values. The accuracy was further deteriorated by the fact that a cross-section library didn't exist for the material at 250°C but the data applied was valid for 600 K (that is 327°C). The measured ITC was -0.781±0.047 c/°C and the simulated one -0.685± 9.776 c/°C, see Table 4. This is the coefficient whose negativity would be the most important for the reactor safety, but as we see from the results, the simulation gave no reliable information about the sign of the coefficient.

Table 4: The simulated and experimental values of isothermal temperature coefficient. ITC was not measured for the temperature interval 20…170 °C

Isothermal temperature coefficient			Benchmark	
Temp	k	Simulated ITC	Measured ITC	(C-E)/E (%)
20	0.99478			
170	0.99198	-0.360 ± 9.951		
250	0.98915	-0.685 ± 9.979	-0.781 ± 0.047	-12.29

5 CONCLUSIONS

The homogeneous model simulated by MCNP provided results some of which are quite well in line with the experimental ones and others not so much. Especially the simulations of the sodium void reactivity and isothermal temperature coefficient illustrated the feature of the Monte Carlo method that its accuracy is strictly limited. Much more computer power or time would be needed. However, even a ten-fold initial neutron population would not diminish the statistical error sufficiently.

Further conclusions may be written when the results of all simulations are available.

ACKNOWLEDGEMENTS

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REFERENCES

[1] Japan's Experimental Fast Reactor JOYO MK-I -core: Sodium-cooled Uranium-Plutonium Mixed Oxide Fueled Fast Core surrounded by UO2 Blanket, International Handbook of Evaluated Reactor Physics Benchmark Experiments, NEA/NSC/DOC(2006)1. OECD/NEA, 2006.