HETEROGENEOUS EFFECT AT FRM-II

YU. V. PETROV and M. S. ONEGIN

Theoretical Department Petersburg Nuclear Physics Institute,
Orlova rosha, 188300 Gatchina - Russia

and

K. BÖNING and M. NUDING

ZBE FRM-II-Bau, Technische Universität München, 85747 Garching - Germany

ABSTRACT

The cores of modern Research Reactors (RR) are characterized by a relatively small volume and a high neutron leakage in order to yield an efficient production of thermal neutrons in the outside moderator. Many RR have fuel elements with aluminum-clad fuel plates cooled by light water. Due to a large difference of the mean free paths of the neutrons in H_2O and Al a relatively large so called "Heterogeneous Effect" (HE) arises in such cores. By definition the HE is the difference between the results for the reactivity obtained from a detailed calculation considering the realistic "heterogeneous" geometry of the core and from a simplified calculation using a "homogeneous" core model in which the fuel plate lattice is homogenized: $\Delta\rho_{HE} = \rho^{het} - \rho^{hom}$. For the first time this negative HE has been found for an aluminum-clad core in the course of calculations performed for the research reactor WWR-M at Gatchina.

In the work as described in this paper the HE was calculated for the FRM-II. The "heterogeneous" calculation has been performed using two different Monte Carlo codes: MCU-RFFI and MCNP-4B with the ENDF/B-VI library. The fuel element of the FRM-II was modeled in detail: the involute shape of all 113 fuel plates, the cladding, the coolant channels etc. were exactly reproduced. The involute shape of each 1.36 mm thick fuel plate was modeled with an accuracy better than 3 μm. Very important is the adequate reproduction of the energy dependence of the Al cross section. The "homogeneous" calculations used for reference were older Monte Carlo code calculations in which the core structure was homogenized from the beginning (i.e. without a special cell calculation). In the calculations performed with the code MCNP the HE is equal to $\Delta\rho_{HE} = -0.5(1)\%$ to $-0.9(1)\%$ depending on the position of the central control rod. For RR with a thicker aluminum cladding of the fuel plates the heterogeneous effect is expected to be larger.

1. Introduction

The fuel elements (FE) of research reactors are heterogeneous. They consist of the fuel-meat, the cladding and water gaps for cooling. For the sake of simplicity of computations it is frequently assumed that all nuclei inside one fuel assembly (FA) are homogeneously mixed. By definition the heterogeneous effect (HE) is the difference between heterogeneous calculations of the reactivity ρ^{het} and the homogeneous ones ρ^{hom}:

$$\Delta\rho_{HE} = \rho^{het} - \rho^{hom} \tag{1}$$

The idea of separating the uranium from a moderator was first formulated by Fermi and Szilard in the summer of 1939 [1]. This separation gives the fast neutrons room to slow down in a moderator escaping the capture in the strong resonances of ^{238}U. Without this *positive HE* the chain reaction in the first large "Natural Uranium–Graphite" Hanford reactors for Plutonium production would have been impossible. The small leakage of neutrons did not change this positive HE.

The research reactors (RR) with high enriched uranium (HEU) fuel are totally different. First of all, the content of ^{238}U in the fuel is small (about 10% or even less) and resonance absorption in ^{238}U is therefore low. Secondly, the RR have small cores with a relatively high neutron leakage (in order to yield an efficient production of thermal neutrons in the surrounding moderator). Due to a difference of the mean free paths of the neutrons in H_2O and in the materials of the FE a significant *negative HE* arises. In contrast to large power reactors the HE at RR is negative rather than positive.

For the first time the negative HE was found during reactivity calculations for critical assemblies of the PIK reactor [2,3]. The PIK fuel elements are cross shaped twisted pins. The pins have a 0.16 mm thick stainless steel cladding and a (UO_2+Cu) meat with a fuel density of 2.1 gU/cm^3 and 6.2(2) gCu/cm^3 in the meat. Data for the PIK FE and FA are presented in Table 1. Benchmark experiments were performed for seven assemblies with fuel pins only (PIK-04) and for eight assemblies with fuel pins in cassettes (PIK-01). The calculations were performed with the Monte Carlo code MCU-RFFI. The discrepancy between experiment and theory (heterogeneous calculations) is rather small: $\rho^{het} \le 0.2\%$. The HE calculated is also shown in Table 1 (row 12). For the highly enriched fuel of PIK the HE is negative: $\Delta\rho_{HE} = -[0.65(8) \ - \ 0.85(14)]\%$.

The negative HE was also established for WWR-M5 FA. The WWR-M5 fuel assemblies used in four benchmark experiments were slightly different from the standard ones [4]. Their data are also shown in Table 1. The meat is an UAl alloy with a thickness of 0.53 mm, the cladding is an Al alloy. The whole plate thickness is 1.25 mm, the width of the water gap is 1.5 mm. The uranium is 90% enriched. The reactivity calculations were performed using the Monte Carlo code MCNP-4B with the ENDF/B-VI library. The calculations were performed for pure materials of the FA, with the 3 empty channels for control rods taken into account. The mean difference in the reactivity between the calculations and the experimental data is shown in Table 1 (row 11). The HE is higher compared to the one of PIK (row 12). For the calculations of the full scale model of the WWR-M reactor at Gatchina (including beryllium reflector, horizontal and vertical beam tubes, ultracold neutron source with shielding etc.) the HE is equal to: $\Delta\rho_{HE} = -0.8\%$. As a conclusion one can expect a significant HE for the FRM-II, too.

2. FRM-II design and parameters

The new 20 MW high-flux reactor FRM-II in Garching near Munich is aimed to be a powerful source of slow neutrons for scientific and applied research [5]. The reactor has a compact core with an active volume of 17.6 l, cooled by light-water and surrounded by a heavy–water reflector in which a high thermal neutron flux is obtained. The fuel is highly enriched uranium (93%) in the form of uranium silicide U_3Si_2 in an aluminum matrix with an aluminum alloy cladding. The 113 fuel plates with an active height of 700 mm and an active width of 62.4 mm form the fuel assembly (FA), located in a tube in the center of the heavy–water reflector tank (Figure 1). As mentioned above, the plates are cooled by light water. The heavy–water reflector tank is located in a shaft inside a light water pool. A cylindrical hafnium control rod filled with aluminum is located in the center of the FA. It is used both as a reactivity regulator–compensator and as a fast shut-down system. A beryllium follower is placed under the hafnium section. In addition five emergency shut-down rods are located in the heavy–water tank.

Table 1.

		Critical assemblies			Reactor
		PIK-04 [2]	PIK-01 [3]	WWR-M5 [4]	FRM-II [7]
1	Meat/Cladding	UO_2+Cu/Fe		U-Al alloy/Al	U_3Si_2+Al/Al
2	Enrichment, %	89.9		90	93
3	U density in the meat, gU/cm^3	2.29		0.783	3/1.5
4	FE wall/clad/meat thickness	1.11/0.143/0.82$^+$		1.25/0.36/0.53	1.36/0.38/0.60

5	Fuel height, H, cm	45		50	70
6	Lattice spacing, a_L, mm	5.3	5.23	–	3.56
7	FA spacing, a_{FA}, cm	–	5.3	3.5	–
8	Meat ratio in cell, ω_M	0.284	0.291	0.173	0.168
9	Mean ^{235}U density in cell, g/cm^3	00.584	0.600	0.122	0.470/0.235
10	H_2O ratio in cell, ω_{H_2O}	0.605	0.594	0.571(4)	0.618
11	Calculated benchmark reactivity, ρ^{het}	0.02(8)±0.13	0.03(7)±0.19	0.31±0.17	–
12	HE: $\Delta\rho_{HE}$, %	–0.65(8)	–0.85(14)	–0.93(11)	–

$^{+)}$ For the cross thickness. The cross diameter is $2(a_i+b)$=4.89 mm.

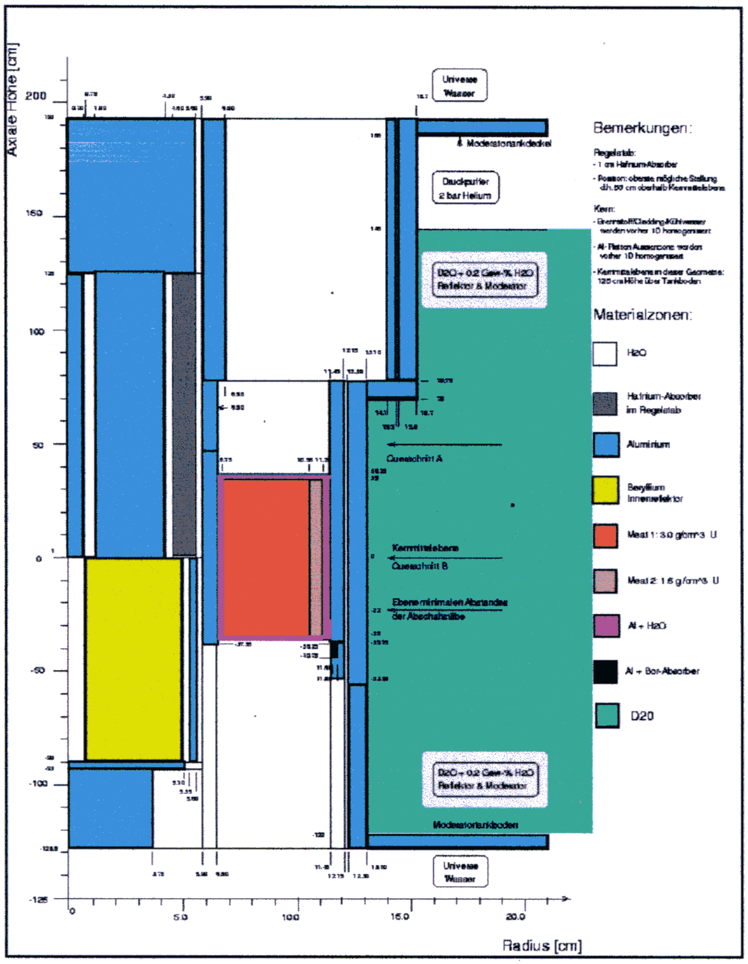
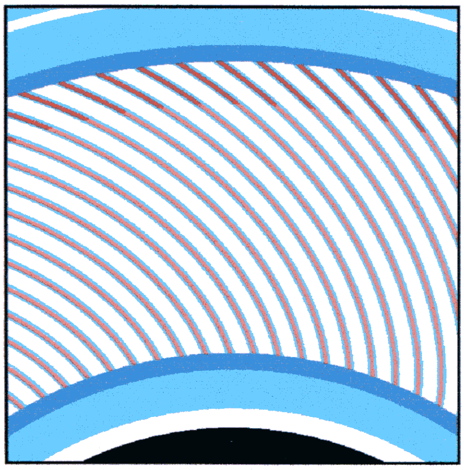


Figure 1: Vertical cross section through a full scale model of the FRM-II core [7].



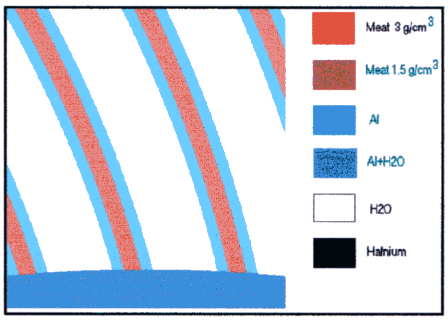


Figure 2: Horizontal section through the FRM-II FA which was drawn by the MCNP code for simulation.

The fuel assembly of the FRM-II comprises 113 involute shaped fuel plates, fixed between two coaxial cylindrical tubes (Figure 2). The profile of the plates $r(\varphi)$ (r is the radius, φ is the angle counted from the internal fixation point) is chosen in such a manner that the gap between them has a constant width at any radius. Together with the coaxial symmetry this provides the constancy of physical and heat characteristics along the core. The specific heat transfer area in the core per FA volume $\xi = S_{th} / V_{br}$ is constant and equals $\xi = 5.62$ cm^2/cm^3. In order to smooth the local power density, the fuel plates have different uranium densities in two radial parts: 3.0 gU/cm^3 between the radii 67.5 mm and 105.6 mm, and 1.5 gU/cm^3 between the radii 105.6 mm and 112.0 mm. The other geometric parameters of the FA are presented in Table 1 [6]. A detailed FRM-II geometry description and the "unreine" (impure) material composition with all isotopes contaminating the core included are given in [7]. These data were used in the calculations described in this paper.

3. Computation methods

The calcualations were performed with the well known Monte Carlo code MCNP-4B with ENDF/B-VI library. MCNP is a general-purpose, continuous-energy, generalized-geometry coupled neutron/photon Monte Carlo code [8]. It solves neutral particle transport problems. The neutron energy region is from 10^{-4} eV to 20 MeV. Capability to calculate k_{eff} eigenvalues for systems containing fissile materials is also a standard feature. Libraries of continuous-energy neutron cross sections and discrete reaction neutron cross sections are available for neutron transport simulation. The cross sections for each reaction and the total cross sections are given on an energy grid which is sufficiently dense for linear-linear interpolation between points to reproduce the evaluated cross sections within a specified tolerance that is generally 1% or less. Depending primarily on the number of resolved resonances for each isotope, the resulting energy grid may contain as many as 22500 points. Angular distributions of scattered neutrons are included in the neutron interaction tables for nonabsorption reactions. MCNP uses the continuous-energy neutron data library MCNPDAT6 which is based on ENDF/B-VI evaluation [9]. Thermal data libraries include chemical binding effects and are available for light and heavy water. The calculated uncertainties of k_{eff} listed are statistical uncertainties at the 1σ level.

Most important for the determination of the HE is an exact representation of the aluminum cross section $\sigma_{Al}(E)$. In Figure 3 the value of $\sigma_{Al}(E)$ is compared with the hydrogen cross section multiplied by the H/Al ratio in water and cladding: $\sigma_{H^*}(E) = \sigma_H(E) \cdot N_H / N_{Al}$. At intermediate energies ($E < 0.5 MeV$) one can see a large difference between these cross sections and minima of $\sigma_{Al}(E)$. As a result in heterogeneous H_2O-Al media a new "cannon" effect arises. At 1 keV the neutron mean free path in aluminum, $\lambda_{Al} \cong 12\, cm$, is much greater than in water: $\lambda_{H_2O} = 0.75\, cm$. The neutron is moving through the aluminum channel like a cannon-ball through the bore of a cannon. Owing to this cannon effect neutrons leak out from heterogeneous media more easily. By homogeneous mixing of the H_2O with Al this heterogeneous cannon effect disappears.

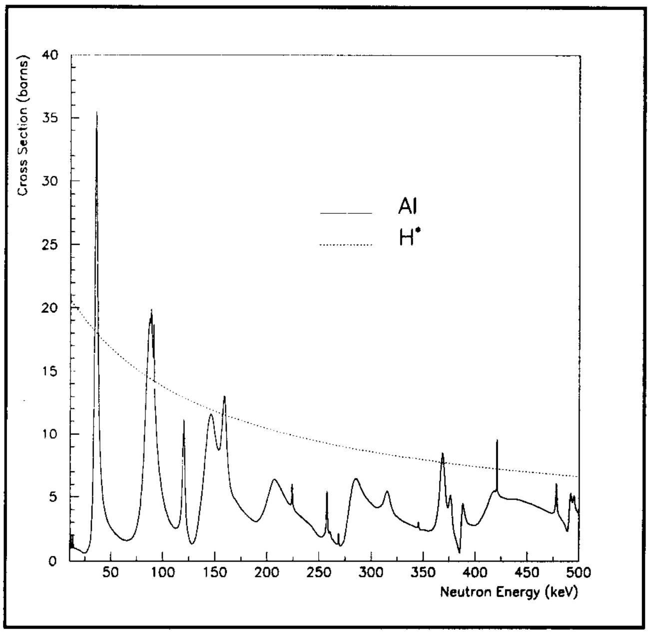


Figure 3: The aluminium cross section $\sigma_{Al}(E)$ and hydrogen cross section multiplied by the H/Al ratio: $\sigma_{H^*}(E)=\sigma_H(E)\cdot N_H/N_{Al}$ [9].

The involute-shaped fuel plates of FRM-II are absent from the a standard geometry elements of MCNP. The equation for the involute plate shape is

$$\varphi = \sqrt{\xi^2 - 1} - \arctan\sqrt{\xi^2 - 1}\,; \quad \xi = r / R_0 \tag{2}$$

$$1 < \xi < 1.74932\,; \quad 2\pi R_0 = N_P a_L \tag{3}$$

Here $N_P = 113$ is the number of plates and a_L=3.56 mm is the lattice spacing. The length of an involute plate in the horizontal plane is equal to l_{in}=0.97431×R_0=62.380 mm. We have approximated Eq.(2) by a parabola and an ellipse belonging to the MCNP standard geometry elements. Optimum parameters of the approximation functions were chosen in such a manner that the mean square deviation from the involute is minimal: $\overline{[\sigma^2]}^{1/2} = 2.9\,\mu m$. The approximation functions were translated into MCNP syntaxes. The equation for the next boundary differs by a shift of the constant angle $\Delta\varphi$. The horizontal through the FRM-II FA is presented in Figure 2, which was drawn by the code MCNP for simulations.

4. Heterogeneous effect at FRM-II

The computations were performed for a fresh, cold (T=300 K) "impure" core without any experimental facilities (EF) in the heavy water reflector. The reactivity ρ was calculated for different positions (STS) of the central control rod (CCR). The reactivity curve for the CCR was calculated for the realistic heterogeneous FA and for the homogeneous case when all nuclei in both fuel regions where totally mixed (Table 2). The difference gives the HE: $\Delta\rho_{HE} = \rho^{het} - p^{\mathrm{hom}}$. The heterogeneous reactivity (ρ^{het}) is systematically lower than the homogeneous one (ρ^{hom}) due to a higher heterogeneous leakage. The heterogeneous effect is significant: $\Delta\rho_{HE} = -[0.5(1) - 0.9(1)]\%$ depending on the CCR position (last column of Table 2). By the insertion of the CCR the thermal neutron current back from the Be follower decreases and the HE increases.

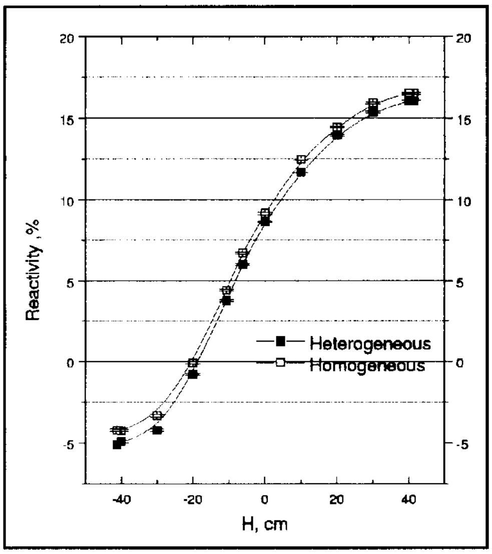


Figure 4: Heterogeneous and homogeneous calculations of the CCR reactivity curve with MCNP and ENDF/B-VI library.

Table 2

	STS, *cm*	k_{eff}^{hom}	k_{eff}^{het}	$\Delta\rho_{HE}$, %
1	41.35	1.1983(7)	1.1912(7)	–0.50(7)
2	40	1.1984(7)	1.1907(7)	–0.54(7)
3	30	1.1897(7)	1.1821(7)	–0.54(7)
4	20	1.1692(7)	1.1624(7)	–0.50(7)
5	10	1.1429(7)	1.1328(7)	–0.78(7)
6	0	1.1009(7)	1.0948(7)	–0.51(8)
7	–6.2	1.0722(7)	1.0638(7)	–0.74(8)
8	–10.67	1.0465(7)	1.0393(7)	–0.66(9)
9	–20	0.9992(7)	0.9923(7)	–0.70(10)
10	–30	0.9680(7)	0.9596(7)	–0.90(10)
11	–40	0.9593(7)	0.9531(7)	–0.68(10)
12	–41.35	0.9596(7)	0.9516(7)	–0.88(10)
$\Delta\rho_{CCR}$, %		20.8(1)	21.1(1)	

The previous homogeneous calculation of the CCR reactivity curve with the MCNP-4B code by W. Gaubatz gives 0.7 % lower values for $k^{hom}(STS)$ [7]. This is due to using the ENDF/B-V library (instead of ENDF/B-VI) and a different cross section for Hf. Earlier the heterogeneous calculations were performed with the Russian Monte Carlo code MCU-RFFI (Kurtchatov Institute code) [10,11]. Using a mean group cross section for aluminum we obtained a much higher HE than given in Table 2. The values for the HE are considerably closer to the data given in Table 2 if we use subgroup cross sections.

The loss of reactivity due to the HE may lead to a loss of core life time. According to a SIEMENS calculation [12] at the end of the reactor cycle a loss of 1 % of reactivity reduces the total reactor cycle life time by about one week. Therefore, the loss of 0.5(1) % of reactivity at *STS*=+41.35 cm due to the HE may reduce the total core life time of 52 days by about 3 to 4 days. Nevertheless, we have to point out that all calculations were performed considering a fresh core. In order to determine the HE at the end of the core life time (i. e. at the point where the reduced excess reactivity becomes important) burnup calculations or calculations considering a burnt FE are necessary. However, it seems that the heterogeneous cannon effect may be important for the total balance of reactivity of the FRM-II. For RR with HEU fuel and fuel plates with thicker aluminum cladding the HE could even be higher.

Acknowledgements

We acknowledge the supply with FRM-II data by W. Gaubatz, the help of V. Petrov in approximating an involute, the calculations for the HE at the reactor WWR-M in Gatchina by E. Sakhnovsky and the assistance of W. von Schlippe in preparing this report.

References

[1] Rhods, R.: "The Making of the Atomic Bomb"; Simons Shuster; New York, 1986. 886p.
[2] Onegin, M.S.; Petrov Yu.V.: "^{234}U and Heterogeneous Effects Contribution to the Reactivity of PIK-04 Assemblies"; Preprint PNPI-2169;Gatchina, 1997; p. 35.
[3] Erykalov, A.; Kotova, L.; Onegin, M.; Petrov, Yu. V.: "Computation of Critical Assemblies PIK-01"; Preprint PNPI-2276; Gatchina, 1998; p. 41.
[4] Petrov, Yu. V.; Erykalov, A. N.; Onegin M. S.: "A Neutron Feasibility Study for Fuel Enrichment Reduction of the PNPI WWR-M Reactor"; Preprint PNPI-2401; Gatchina, 2000; p. 51.
[5] Böning, K.; Nuding, M.: "Die Neutronenquelle FRM-II"; *In:* Neue Forschungs-Neutronenquelle Garching; ZBE FRM-II-Bau, Technische Universität München, Garching, 1998; p. 119.
[6] Böning, K.; Gallmeier, F.; Gaubatz, W.; Kuzminov, V.; Nuding, M.; Onegin, M.; Petrov,Yu. V.; Smirnova, T.: "Response of the Reactor FRM-II to a Postulated Overcriticality Taking the Control Rods into Account"; Preprint PNPI-2248; Gatchina, 1998; p. 70.
[7] Gaubatz, W.: "Rechnerische Optimierung der sekundären Neutronenquellen des neuen Münchner Forschungsreaktors FRM-II"; Dissertation; ZBE FRM-II-Bau, Technische Universität München, Garching, 1998; p. 121.
[8] Briesmeister, J.: "MCNP - A Ceneral Monte Carlo Code for Neutron and Photon Transport"; LA-7396-M, 1986; LA-12625 M, 1993.
[9] Rose, P. F.; Dunford, C. L.: "ENDF-102 Data Format and Procedures for the Evaluated Nuclear Data file ENDF-VI"; US National Nuclear Data Center; BNL, Upton, NY, USA-11973.
[10] Gomin, E. A.; Gurevich, M. I.; Maiorov, L. V.; Marin, C. B.: "The MCU-3 code for Monte Carlo calculations of nuclear reactor neutronics"; Vol. 1. Discription of an application and manual for users; Preprint IAE-5772/5; Moscow, 1994.
[11] Abagian, L. P.;Glushkov, A. E.; Gomin, E. A.; Kalugin, M. A.; Maiorov, L. V.; Yudkevitch, M. C.: "The MCU-3 code for Monte Carlo calculations of nuclear reactor neutronics"; Vol. 2. Neutron cross Sections; Preprint IAE-5777/5, Moscow, 1994.
[12] Sicherheitsbericht: Neutronenquelle München FRM-II; Technische Universität Münchnen und Siemens AG; 1993.