ANALYSIS OF NEUTRON FLUX DISTRIBUTION FOR GAS COOLED REACTOR ALLEGRO WITH MOX FUEL

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1. Introduction

The paper is dedicated to the neutron flux calculation for the fast neutron demonstrator unit ALLEGRO. There are more applicable designs of the core and the paper is concerned in ESNII+ 75 MW_{th} ALLEGRO reactor conceptual design.This fast spectrum reactor is helium-cooled system.According to the plan,the reactor may be sited in the Slovakia.

PARTISN is neutron flux calculation softwarewhich was used in this case. It isa modular computer package program designed to solve the time-independent or dependent mutligroup discrete ordinates form of the Boltzmann transport equation in several different geometries. Benchmark testing of the calculation software is introduced in the separate paragraph. Subsequently, the homogenization used to transform heterogeneous model of the ALLEGRO core to RZ form is described. It was necessary to develop a code, which is able to translate binary files into the ASCII format and is able to plot pictures by usage ofgraphical editor GNUPLOT. The whole program code was developed in C/C++ compiler.

2. Benchmark testing

Benchmark testing was performed for the purpose of PARTISN calculation process validation [1,2]. Three cross section libraries were used to k_{eff} calculation, namely: ZZ-KAFAX-E70 (based on ENDF/B-VII.0), ZZ-KAFAX-F31 (based on JEFF-2.2) and ZZ-KAFAX-J33 (based on JENDL-3.3). These libraries were originally generated for the KALIMER (Korea Advanced Liquid Metal Reactor) core analyses.Each library is dedicated for calculation offast reactor system. The results of Benchmark testing are shown in Fig.1. Results of calculation obtained by usage of cross section library ZZ-KAFAX-E70 are the most corresponding with the benchmark k_{eff} with particular uncertainty.

The most accurate calculation was performed on the nuclear assembly called MIX-COMP-FAST-002 (hereinafter ZPR-6/7). This cylindrical assembly use mixed (Pu, U)-Oxide fuel and is equipped with a central high ^{240}Pu zone.This assembly was constructed to support fast reactor development. The ZPR-6/7 Benchmark Assembly had a very simple core unit cell assembled from plates of depleted uranium, sodium, iron oxide, U_3O_8, and plutonium. The ZPR-6/7 core cell-average composition is typical of the interior region of liquid-metal fast breeder reactor (LMFBR) cores anticipated in design work in 1970.

On the other hand, the most inaccurate result is obtained on the nuclear assembly called IEU-MET-FAST-005. The critical assembly is a sphere with a core of ^{235}U(enrichment 36%) metal having a central cavity and steel reflector. In the calculation, the assembly is represented by a simplified model with the core homogenous in composition and a two-layer steel reflector [1]. The differences between calculated results and benchmark k_{eff} are unacceptable in this case. The deviation of almost 2800 pcm need to be further investigated.

The most probable reason for this is the incorrectly created material or geometry model in PARTISN calculation.

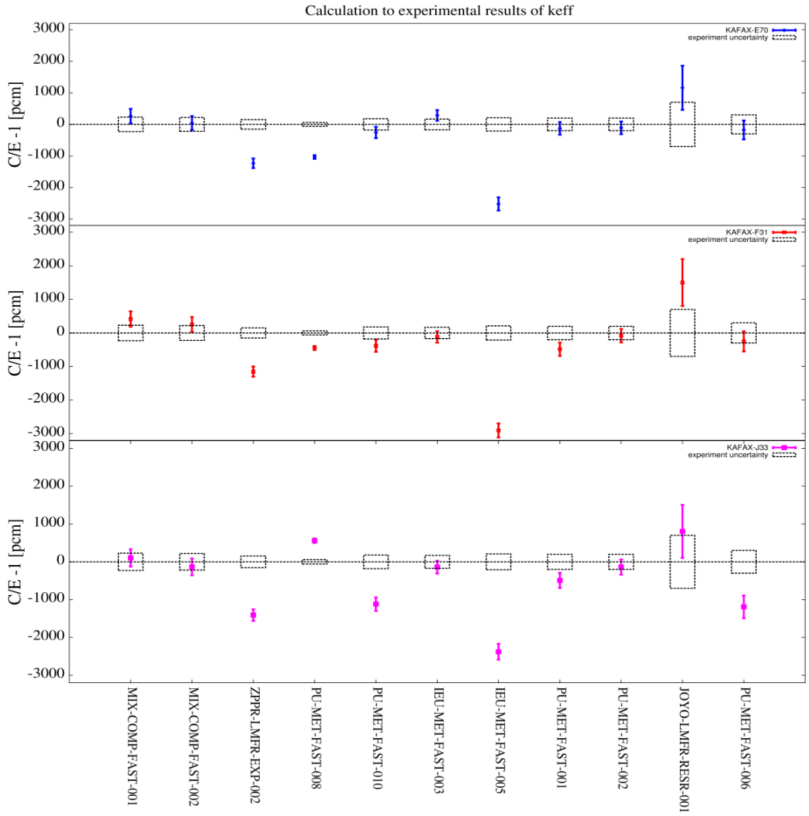


Fig.1:*Comparison of calculation results with Benchmark models.*

3. The homogenization of the ALLEGRO MOX core

ALLEGRO core composition includes 81 fuel assemblies (each with 169 fuel pins). Additionally, the core contains 6 experimental subassemblies (STEEL assembly), 4 Diverse Shutdown Devices (DSD) and 6 Control and Shutdown Devices (CSD). The geometry is described in Fig.2.

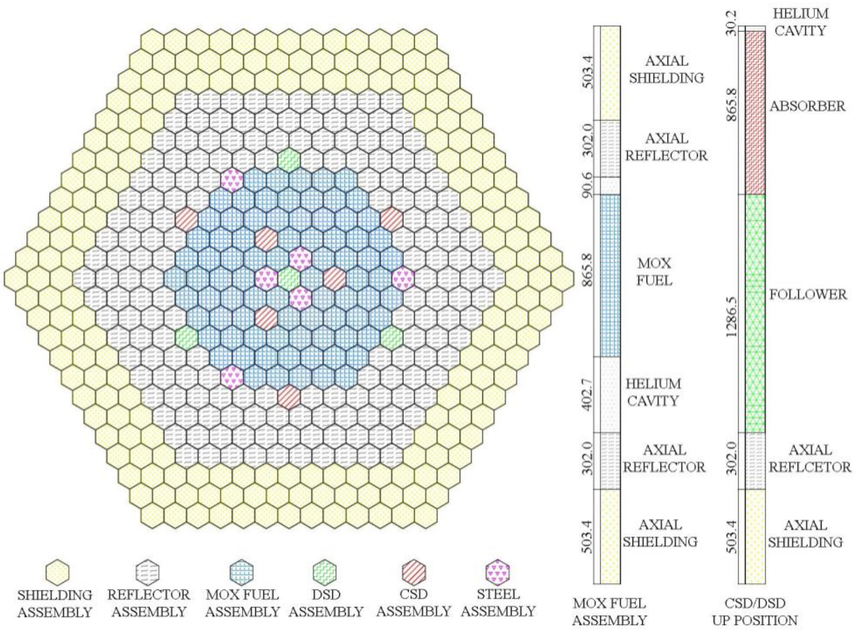


Fig.2:*Geometry of ALLEGRO MOX core.*

STEEL, Radial Reflector and Radial Shield assemblies are homogenous in vertical axis. CSD and DSD assembly contains absorber in the form of B_4C with 70% enrichment of ^{10}B.PuO_2volume content of fuel assembly is 25.5%.

The homogenization was performed in order to simplify heterogeneous geometry of ALLEGRO core. All control rods were placed in the upper position.The new homogenous geometric model was dependent on two variables: r-radius, z-height.The geometry with the new calculated dimensions is shown inFig.3. The radial symmetry is illustrated by hachures on the left part of Fig.3. The hachures represent reflective boundary condition. Columns expressed by *capital letters* represent cylinders with different radius showed on the bottom of the picture. Rows expressed by *number* representdifferent height of the core with dimensions located on the right side of the picture.

The main adopted idea of homogenization was to keep the same volume of core materials and the same volume fractions of nuclides in the new model. During the geometry transformation from hexagon to cylinder, the overall radius of core is reduced.

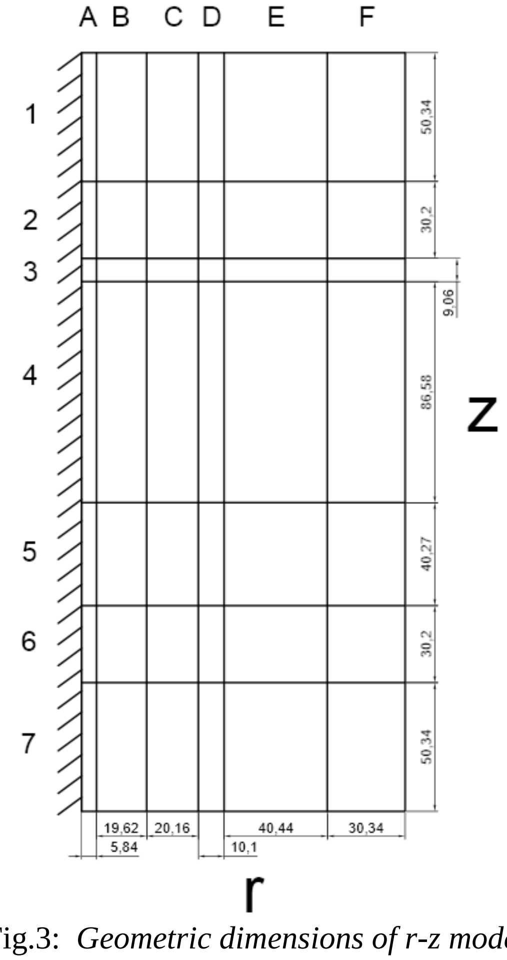


Fig.3: *Geometric dimensions of r-z model.*

The reason is that cylinder is more compact than hexagon. If core radius is smaller, the average distances between the nuclides in core are also smaller. Another effect is the change of leakage term in neutron balance equation. In fast reactor cores, the negative change of core radius causes decrease in neutron leakage term and consequently k_{eff} of system rises.In other words, the k_{eff} should increase slightly with these geometry changes.

Each cylinder is homogenously mixed up from several assemblies in order to keep the same volume fractions of nuclides in the core.

The composition of the Cylinder A is one DSD absorber assembly. A1 node is absorber part of the DSD assembly. 3.02 cm high helium cavity is on the top. This additional ^4He is homogenously mixed up within the absorber. Also nodes A2 and A3 are absorbers (in this case with no additional ^4He). A4 and A5 nodes represent follower part. A6 axial reflector and A7 axial shield.

The Cylinder B is composed from 18 assemblies: 12 MOX fuel assemblies, 3 CSD absorber assemblies (nuclide composition of CSD assembly is similar to the DSD assembly) and 3 steel assemblies. Steel assembly is filled with ^4He and surrounded by a stainless steel wrapper. The B1 node is a homogenous mixture of: 12 axial shields of MOX assembly, 3 parts of STEEL assembly and 3 absorber parts of CSD assembly (similarly as in A1 node, the helium cavity is mixed in the absorber). B2 is a mixture of: 12 axial reflector parts of MOX assembly, 3 parts of STEEL assembly and 3 absorber parts of CSD assembly (as in the case A2 node, there is no additional ^4He from helium cavity).

B3 node is composed of: 12 helium cavities of MOX fuel assembly, 3 parts STEEL assembly and 3 absorber parts of CSD assembly. B4 node: 12 MOX fuel parts, 3 parts of STEEL assembly, 3 follower parts of CSD assembly. B5 node: 12 helium cavities of MOX fuel assembly, 3 parts of STEEL assembly and 3 follower parts of CSD assembly. B6 node: 15 axial reflector parts (12 from MOX fuel assembly and 3 from CSD absorber assembly), and 3 parts of STEEL assembly. B7 node: 15 axial shield parts (12 from MOX fuel assembly and 3 from CSD absorber assembly), and 3 parts of STEEL assembly. Cylinder C is composed of 42 MOX fuel assemblies. C1 and C7 nodes are composed of: 42 axial shield parts. C2 and C6nodes: 42 axial reflector parts. C3 and C5 nodes: 42 helium cavity parts. C4 node: 42 MOX fuel parts. The Cylinder D is composed of 30 assemblies: 27 MOX fuel assemblies and 3 steel assemblies. D1 and D7 nodes are composed from: 27 axial shield parts and 3 parts of STEEL assembly. D2 and D6 nodes: 27 axial reflector parts and 3 parts of STEEL assembly. D3 and D5nodes: 27 helium cavity parts and 3 parts of STEEL assembly (the nuclide composition of helium cavity and STEEL assembly is the same). D4 node: 27 MOX fuel parts and 3 parts of STEEL assembly. The composition of the Cylinder E is: 174 radial reflector assemblies, 3 CSD absorber assemblies and 3 DSD absorber assemblies. The E1 node is a homogenous mixture of: 174 radial reflector parts and 6 absorber parts of CSD/DSD assembly with additional ^{4}He. E2 and E3 nodes: 174 radial reflector parts and 6 absorber parts of CSD/DSD assemblywithout additional ^{4}He. E4 and E5 nodes: 174 radial reflector parts and 6 follower parts of CSD/DSD assembly. E6 node is mixture of: 174 radial reflector parts and 6 axial reflector parts. E7 node: 174 radial reflector parts and 6 axial shield parts. The composition of the Cylinder F is 198 radial shield assemblies. The nuclide composition is the same among each node of cylinder.

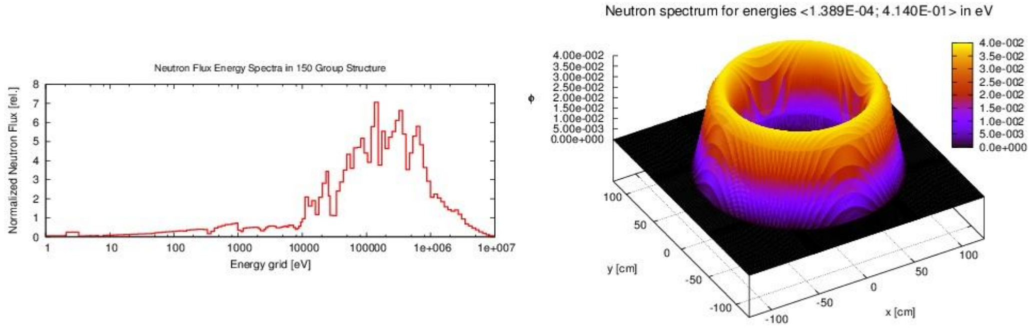
4. Results

The calculation process used 4^{th} degree of Lagrange polynomials withBell-Hansen-Sandmeier (BHS) transport approximation. BHS makes attempt to correct for anisotropy in the scattering matrix and is especially effective for forward-peaked scattering [3,4]. This transport correction was used for the purpose of energy group collapsing in the future. The results in the comparison with other heterogeneous calculations, performed at INPE, are shown in Tab.1.

Tab. 1.*Comparison of the results.*

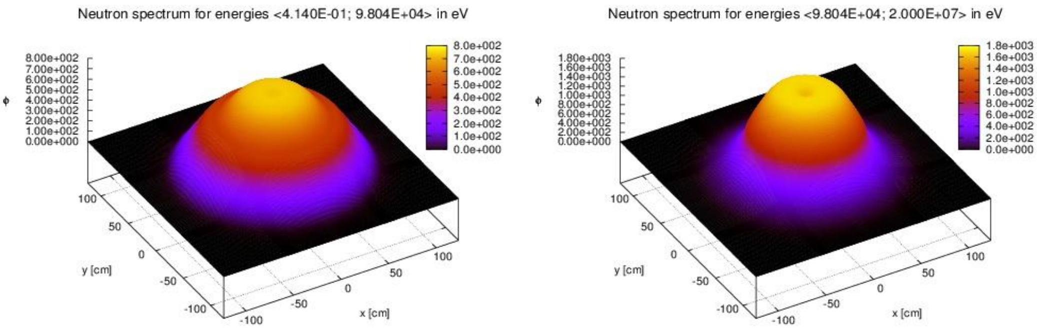
Computer code	**k_{eff}**	**Computer code**	**k_{eff}**
PARTISN ZZ-KAFAX-E70	1.01573852	MCNP continuous	1.011390±0.00008
PARTISN ZZ-KAFAX-F31	1.01904224	SCALE continuous	1.013030±0.00010
PARTISN ZZ-KAFAX-J33	1.01463201	SCALE 238 group	1.026111±0.00004

As mentioned in chapter 3, k_{eff} calculated by PARTISN is slightly higher than the k_{eff} calculated by heterogeneous codes. Neutron flux energy spectra, shown in Fig.4-a), support the fact that ALLEGRO is fast nuclear reactor.The mean energy of the spectra is 4.97E+05eV and most probable energy is 1.34E+05eV. Neutron flux distribution for thermalenergy spectrum is shown in Fig.4-b) andfor resonant in Fig.5-a)in the horizontal cut of the core (the height of162.37 cm).It is possible to seein Fig.4-b) that the neutrons are moderated into the thermal spectrum out of the core in the radial reflector region. These neutrons are reflected back to the core and they induce fission of ^{235}U which is needed mainly in the first core startup. Neutron flux in resonant spectrum is almost doubled in comparison with thermal spectrum. For instance ^{239}Pu is breading from ^{238}U in this energy spectrum andthe peak of this neutron flux spatial distribution is located in the middle of the core. Fast neutrons induce fission mainly on all isotopes ofPu andits spatial neutron flux distribution is shown in Fig.5-b).



a) Energy spectrab) Thermal energy

Fig.4: *Neutron flux spatial distributionin the horizontal cut of the coreand energy spectra.*



a) Resonant energy b) Fast energy

Fig.5: *Neutron flux spatial distribution in the horizontal cut of the core.*

5. Conclusion

Except of one case, which will be further inspected, the used cross section libraries are consistent after Benchmark testing. These libraries are available for calculation of fast spectra gas cooled reactors. Results of multiplication coefficient from PARTISN are also acceptable in accordance with results of other heterogeneous codes. The developed code, called *BBK*, is able to automatically visualize the spatial neutron flux distribution.Next stage lays incomparison of neutron flux energy spectra with heterogeneous codes and to prepare cross section libraries for diffusion codes. These deterministic codes are highly suitable for fast neutron flux spatial distribution calculation and successful usage of these codes shows theknowledge and familiarity with neutron physics.

Acknowledgement

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References:

[1] International Handbook of Evaluated Criticality Safety Benchmark Experiments,NEA Nuclear Science Committee, (2010).
[2] R. E. Alcouffe, et al.: PARTISN A Time-Dependent, Parallel Neutral Particle Transport Code System, Los Alamos National Laboratory, New Mexico USA (2008)
[3] RSICC Peripheral Shielding Routine Collection, TRANSX 2.15, Los Alamos National Laboratory, New Mexico USA (1995)
[4] P. Reuss: Neutron Physics, EDP Sciences, Ulis France (2008)