INVESTIGATION OF THE ALLEGRO ACTIVE CORE USING THE DIFFUSION DIF3D CODE

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1. Introduction

Presented work deals with active zone calculations of the ALLEGRO gas cooled fast reactor by means of the diffusion DIF3D code. Being a demonstrator unit for GFR 2400, this reactor is not meant to produce electricity. It has been designed to prove and test key technologies and materials needed for GEN IV GFR. When built, it will be the first reactor utilizing both fast neutron spectrum and high thermal efficiency and a key step towards commercially viable gas cooled fast reactors. According to the Sustainable Nuclear Energy Technology Platform, the GFR technology has the potential to increase sustainability by minimizing waste [1]. Therefore it has been chosen as one of the six GEN IV concepts. The reactor is being developed in European-wide cooperation under the European Commission GoFastR project [2].

With internal pressure of 70 bar and fuel temperature reaching almost 900°C, core materials will have to withstand extreme conditions for long periods of time. For the new material composition to be used in the active core, reactivity changes have to be assessed. Deterministic codes such as DIF3D and PARTISN are able to give power density maps and flux density maps, enabling region-wise search for criticality problems. These codes solve the transport equation by means of numerical methods, giving exact answers to investigated problems, not burdened with statistical uncertainties, which is one of the biggest disadvantages of stochastic codes, such as MCNP.

1.1 Description of calculation codes and cross-section libraries

In the performed calculations, two different libraries were used with the hexagonal model of the active zone in DIF3D code. The DIF3D is a deterministic code designed mainly for solving fast reactor problems. The nodal option of DIF3D solves the multi group steady-state neutron diffusion equation in two- and three-dimensional hexagonal and Cartesian geometries. Eigen value, adjoint, fixed source and criticality search problems are permitted as are anisotropic diffusion coefficients. Flux and power density maps by mesh cell and region-wise balance integrals are provided [3].

The DIF3D code requires cross section libraries including tabulated multi group data of the most important reactions as an input file, with ISOTXS being a preferred file format. To create the effective macroscopic cross-section (XS) libraries from MATXS microscopic cross-section libraries TRANSX [4] was used. The code has been developed for fission, fusion, and shielding applications at Los Alamos and just later; extensions to handle heterogeneous self-shielding problems for fast reactors were added. The TRANSX code serves for interfacing XS libraries to transport tables compatible with many discrete-ordinates and diffusion codes.

The PARTISN code was used to collapse the energy group structure from 150/620 to 25 energy groups. It uses the same nuclear data libraries; however, in comparison to DIF3D, it works with Cartesian and cylindrical three-dimensional geometries and does not require simplifications in the angular dependence of the transport equation. It numerically solves the multi-group form of the Boltzmann transport equation and it employs the discrete-ordinates form of approximation for treating the angular variation of the particle distribution [5].

2. The computational approach

2.1 Core model

The hexagonal model of the active zone was created according to the scheme in the figure 1. In the DIF3D model, vertical mesh step was set to 5 mm and dimension of hexagon across flats to 111.258 mm. All the other dimensions were set according to ALLEGRO specifications [6].

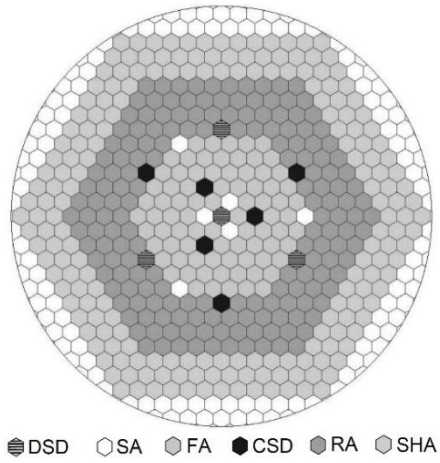

Fig.1:*Hexagonal active corelayout of the DIF3D model*

In the picture, DSD is control rod, SA stands for steel assembly, FA for fuel assembly, CSD is control and shutdown device, RA reflector assembly and SHA shield assembly. The figure was generated using SCALE6. In the picture, hexagonal mesh continues beyond the outer boundary of the active core, determined by shielding, however only the active core has been modelled. In the DIF3D model, interpolated vacuum boundary conditions were used.

2.2 Computational scheme

In the Fig.2, the computational scheme used for calculating the effective neutron multiplication factor using the first investigated library, ZZ KAFAX E70 [7], is shown. The library is in MATXS format, required by the TRANSX code and is retrievable from the OECD Nuclear Energy Agency webpage. First, using material composition from the ALLEGRO specifications, an input file for the TRANSX code was created. The input file requires isotopic composition and temperature of homogenized sets of materials. The material composition of the subassemblies was homogenized into 8 sets of materials, namely, fuel, axial and radial reflector, axial and radial shield, absorber, follower and helium cavity. Each of the sets comprises all the materials used in the corresponding section of the subassembly. The homogenization was done using the volume fraction of each isotope in the used materials with respect to the total volume of the subassembly. When the input file is processed,

TRANSX calls the ZZ KAFAX E70 library, where 150 group neutron and 12 group photon cross-section data are stored, and gives a set of macroscopic cross section libraries in the ISOTXS format [4]. The output file was created using 150 energy group structure and Legendre polynomial sequence of the 0-th order. Resulting ISOTXS was further processed through a model of the active core, created in the DIF3D code.

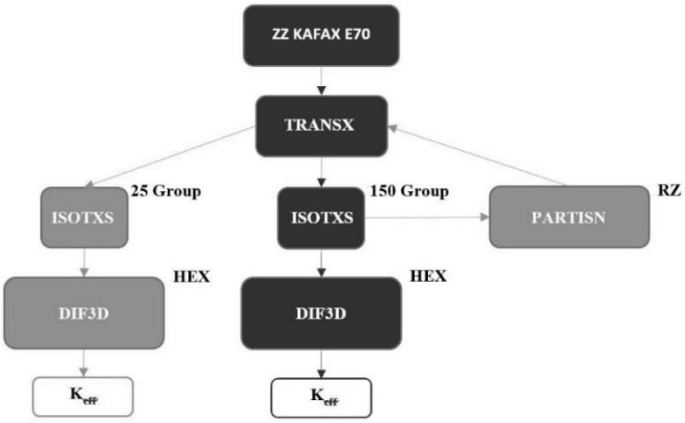

Fig.2:*The computational scheme using ZZ KAFAX E70, where RZ and HEX refer to geometry*

To speed up the DIF3D calculation and to account for transport effects it is sufficient to collapse the group structure using region dependent transport RZFLUX. The 25 group cross section library was created by processing the 150 group ISOTXS with the PARTISN code and collapsing into 25 groups. It is important to mention that the DIF3D model uses 3D hexagonal geometry, while the PARTISN model 2D cylindrical RZ geometry.

The same computational scheme was used for SBJ620GE71 library [8], which was prepared using the NJOY99 code, as shown in the Fig. 3.

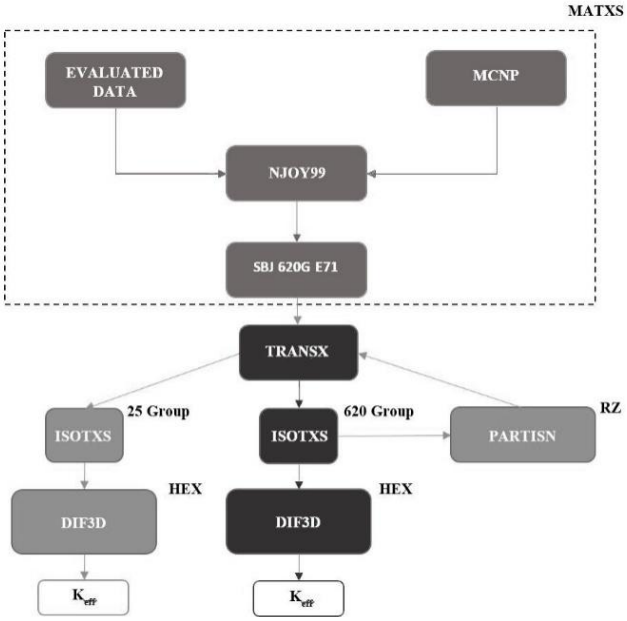

Fig.3:*The computational scheme using ENDF/B-VII.1 evaluated data, where RZ and HEX refer to geometry*

ENDF/B-VII.1 evaluated data were used and the MCNP calculated average neutron spectrum as a weighing function. Using the TRANSX code, 620 group isotope-wise ISOTXS was prepared.

3. Results

Comparison of the effective neutron multiplication factors calculated with the two different libraries, each with two different energy groups, can be seen in the Tab. 1. The values are calculated for the case, when all the control rods are in the upper parking position "All-Up". Computational scheme and hexagonal model described in the previous section were used.

Tab. 1.*Effective neutron multiplication factor results table*

Library	ZZ KAFAX E70		SBJ 620G E71	
Groups	25	150	25	620
K_{eff}	1.011253	1.008142	1.016317	1.008272

In the Table 2, weights of control rods calculated using 25 energy groups are displayed. The deviations of the DIF3D values fromthe reference MCNP5 calculations are shown. For single CRs this discrepancy did not exceed 2 %. This accuracy is sufficient, since in MCNP5 the heterogeneous CR design was investigated, while in DIF3D homogeneous compositions were used. The discrepancy between the two DIF3D calculations can be considered negligible.

Tab. 2.*The weights for different control rods in comparison to reference values*

Control rod ID	ZZ KAFAX E70		SBJ 620G E71	
	ρ [pcm]	Δ [pcm]	ρ [pcm]	Δ [pcm]
AD	11565.4	1169.2	11800.6	934.0
CSD3.1	2172.4	137.6	2199.6	110.4
CSD7.4	436.5	6.4	451.5	8.6
DSD1.1	2342.3	32.3	2397.1	246.8
DSD7.10	451.5	3.7	451.5	3.7

In the Tab. AD stands for the case with all control rods inserted into to core (All-Down) and ID refers to position with ring number and position in the ring separated by dot product.

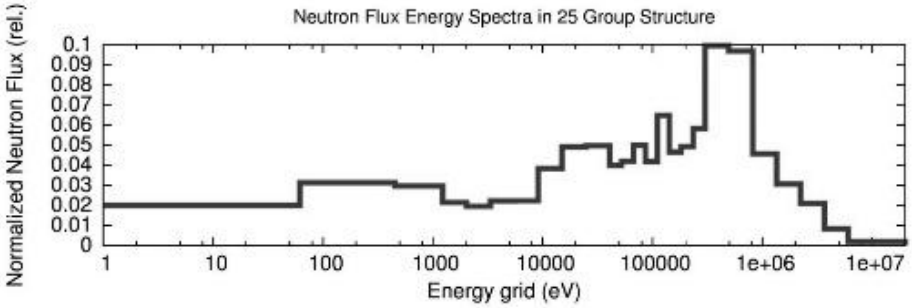

Fig. 4:*Normalized neutron flux energy spectrum using 25 energy groups*

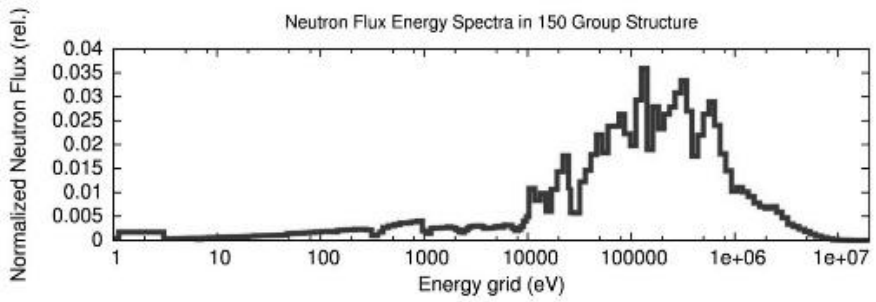

Fig. 5:*Normalized neutron flux energy spectrum using 150 energy groups*

The neutron flux energy dependences in fuel are shown in the figures above. Characteristics using 25 energy group input (Fig. 4) and 150 energy group input (Fig. 5) using the KAFAX library are displayed. From the figures, it can be seen that the appearance of the neutron spectra strongly depend on the energy group structure.

4. Conclusions

In this study the neutronic performance of the ALLEGRO reactor core using DIF3D model and two different libraries was investigated. As reference, values calculated in MCNP5 were used. The computational time and demands on the hardware increase rapidly using more energy groups. Using 620 energy groups, the computational time of the DIF3D ALLEGRO active core model however seems still acceptable and higher energy resolution is able to give more accurate results. In comparison to MCNP5, which requires days, DIF3D calculations preformed on the same hardware require around 10 seconds. From the computational point of view, it can be said, that performing the ALLEGRO core calculations by DIF3D is advantageous, since the precision is comparable with that of MCNP5.

Acknowledgement

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References:

[1] M.B. Chadwick, P. Obložinský and M. Herman: *ENDF/B-VII.0*: *Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology*, Nuclear Data Sheets**107**,p. 2931(2006).

[2] C. Poette: *ALLEGRO 75 MW Cores Definition at Start of GOFASTR – GoFastRDeliverable,* CEA(2010).

[3] ORNL: *DIF3D: Code System Using Variational Nodal Methods and Finite Difference Methods to Solve Neutron Diffusion and Transport Theory Problems*, RSIC(2011).

[4] R. E. MacFarlane et al.: *TRANSX 2: A code for Interfacing MATXS Cross-Section Libraries to Nuclear Transport Codes*, Los Alamos National Laboratory, LA-12312-MS, Los Alamos, USA (1992).

[5] R. E. Alcouffe, et al.: *PARTISN: A Time-Dependent, Parallel Neutral Particle Transport Code System*, LA-UR-08-07258, Los Alamos National Laboratory(2008).

[6] E. Temesvári, ESNII+: *ALLEGRO Core Specifications*, internal deliverable D6.6.1-2(2014)
[7] D. H. Kim, C. S. Gil and Y. O. Lee,: *ZZ KAFAX-E70, 150 and 12 Groups Cross Section Library in MATXS Format based on ENDF/B-VII.0 for Fast Reactors*, Nuclear Data Evaluation Lab., Korea Atomic Energy Research Institute, Daejeon, Korea (2008).
[8] J. Haščík, Š. Čerba, , J. Lüley and B. Vrban: *Neutronic characterization of the GFR demonstrator*, in Proceedings of the 7th International Conference on Modelling and Simulation in Nuclear Science and Engineering (7ICMSNSE),Ottawa, Ontario, Canada, (2015).