THERMAL NEUTRON DISTRIBUTION MEASUREMENT AT CORE OF THE LVR-15 REACTOR

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ABSTRACT

The LVR-15 reactor is a light water research type which is situated in Rez near Prague. The maximum thermal power is 10 MW. The LVR-15 reactor is used mostly for irradiation of material specimens for a given period with appropriate neutron fluence rate. It is important for such experiments to know neutron fluence and energy spectrum in the sample. The paper describes measurement of thermal neutron distribution in four vertical channels in core of the LVR-15 reactor. The activation method with Cu foils was used for the measurement. The mean power during the irradiation was 0.1 MW. The gamma activities were measured with spectrometric assembly including the HPGe detector. Nearly all the data were loaded automatically without any manual typing which guarantees the high reliability of final results. The results of the measurement have been compared with data calculated with three computer codes - with a diffusion nodal program NODER in a four group approximation, with the general purpose Monte Carlo radiation transport code MCNP-4B and the DLC-189 library, and with the two-dimensional transport code DORT. The measurement and calculation was applied for reactor core configuration which was specially designed for boron neutron capture therapy experiments in June of 1999.

1. INTRODUCTION

The reactor LVR-15 is a light water research type which is situated in Rez near Prague. The reactor LVR-15 has been operated after the second reconstruction of the original WWR-S reactor which had been operated since 1957. The first criticality occurred in September 1957 and was the first achieved in Central Europe. Original power was 2 MW. The first modernisation was made from 1974 to 1975 when the original fuel was changed to IRT-2M fuel (concentration ^{235}U of 80%). The second modernisation was realised between 1985 and 1990 when the reactor vessel was exchanged. The original Al alloy vessel was replaced by a stainless steel one which was produced by Skoda Plzen machinery. At this time also control system and other systems were exchanged. At present the transition on IRT-2M fuel of Russian production with enrichment of 36% has been finished. In the reactor core there are usually from 28 to 31 fuel elements with the total mass about 5 kg of ^{235}U. The maximum thermal power is 10 MW. Reactor is cooled by demineralized water.

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The paper describes measurement of thermal neutron distribution in core of the reactor LVR-15. The results of the measurement have been compared with data calculated by three computer codes - NODER, MCNP4B and DORT programs.

2. DESCRIPTION OF THE EXPERIMENTAL WORK

The activation method, the reaction $^{63}Cu\ (n,\gamma)\ ^{64}Cu$, was used for measurement of thermal neutron distribution in core of the reactor LVR-15. Thus the material of the monitors was 99.99% Cu, the form was disc with diameter of 4 mm and a thickness of 0.1 mm. As a support 80 cm long Al sticks were used and the step between monitors was 5 cm.

The activation monitors were weighted with the Mettler Toledo balance type AT261. The readability of the measured values is 0.01 mg while the typical mass of the monitor is 14 mg. The value of the measured mass is automatically stored in the Microsoft Excel data file.

Four sticks with monitors were irradiated to determine vertical profiles in four different channels in the reactor. During the monitors irradiation the values of reactor power were measured and stored.

The gamma activities were measured by spectrometric assembly (Canberra) with the HPGe detector. The spectrometric assembly had the relative efficiency of 18% and FWHM=1.8 keV for energy of 1332 keV. The detector is placed in a shielding box with 5 cm thick lead walls. The calibration of the detector and the method of the measurement are in accordance with ASTM E 181-82 [1]. An automatic sample changer was used for changing of samples.

All the measured data which are necessary for the reaction rate evaluation - the mass of monitors, the activities and their errors, the time dependence of power during irradiation - are stored in the previously mentioned Excel file. Nearly all the data are loaded automatically without any manual typing which guarantees the high reliability of final results. The reaction rate are calculated by a prepared Excel Visual Basic for Application program [5].

3. CALCULATIONS

The thermal neutron distribution in the four positions were calculated with three computer codes.

The first set of calculations, used for the determination of the neutron fluence in observed regions, was performed by a diffusion nodal program NODER in a four group approximation. Outputs from performed core calculation (neutron source density, thermal power distribution) were also used as inputs for detailed transport and Monte Carlo calculations.

The second set of calculations was performed with the general purpose Monte Carlo radiation transport code MCNP-4B [2] and the DLC-189 library on a model of the thermal neutron distribution.

The third set of calculations was performed with the two-dimensional transport code DORT [3]. The three-dimensional neutron fluxes were substituted by the three-dimensional fluxes rate synthesis, in calculations the BUGLE96 multigroup data library was used [4]. The neutron fission density distribution in the core is based on the NODER calculation.

3. RESULTS

3.1 CONFIGURATION OF THE REACTOR CORE

The measurement and calculation was applied for reactor core configuration which was specially designed for boron neutron capture therapy (BNCT) experiments in June of 1999. A horizontal section of the reactor core can be seen in the Fig. 1. The BNCT facility uses fast neutrons which escape the reactor core in direction of the thermal column (Fig.1). The pitch of fuel elements of a square section arranged into a rectangular lattice is 7.15 x 7.15 cm. Water channel between every two fuel units for cooling is 0.45 cm. The core (B2 - G7) is usually surrounded by Be reflector units placed in the row No.8 and rows No.9-10 are filled with water. To enable the core neutrons to reach to the BNCT facility some type of fission converter was installed at C8 - F10 positions.

The measurement and calculation were made for channels C6, D7, C10 and D10. On next graphs on axis Position (z-axis) the value -30 cm corresponds to the bottom of the reactor core, value 0 to the centre of the core and value +30 cm to the top of the core.

Thermal column

⇧

	A	B	C	D	E	F	G	H	
10	Water	Be	**Monitors** Fuel	**Monitors** Fuel	Fuel	Fuel	Be	Water	10
9	Water	Water	Air	Air	Air	Air	Water	Water	9
8	Water	Water	Air	Air	Air	Air	Be	Water	8
7	Be	Be	Fuel	**Monitors** Fuel	Fuel	Fuel	Be	Be	7
6	Be	Fuel	**Monitors** Fuel	C. rod Fuel	C. rod Fuel	Fuel	Fuel	Be	6
5	Be	Fuel	C. rod Fuel	C. rod Fuel	C. rod Fuel	C. rod Fuel	Fuel	I. Ch.	5
4	Be	Fuel	C. rod Fuel	C. rod Fuel	C. rod Fuel	C. rod Fuel	Fuel	Be	4
3	Be	Fuel	Water	Fuel	C. rod Fuel	Fuel	C. rod Fuel	Be	3
2	Be	Fuel	Fuel	Fuel	I. Ch.	I. Ch.	Fuel	I. Ch.	2
1	Water	Be	Be	Be	I. Ch.	I. Ch.	I. Ch.	I. Ch.	1
	A	B	C	D	E	F	G	H	

Be	Beryllium
C. rod	Control rod
I. Ch.	Irradiation channel

Figure 1. LVR-15 reactor core on 24. June 1999.

3.2 MEASUREMENT WITH ACTIVATION METHOD

The time of irradiation of monitors in the 4 stick was about 2.5 hours, the mean power about 0.1 MW. For determination of ^{64}Cu activity both 511.0 keV and 1345.8 keV energy lines ware used. The range of measured activities is 8 x 10^5 Bq to 7 x 10^7 Bq. The typical error of activity values was 2 % on 1σ STD level. The range of calculated reaction rates is 7 x 10^{-14} s^{-1} to 7 x 10^{-12} s^{-1}. The measured reaction rates were transformed into the thermal neutron fluence rate using an effective thermal cross section σ_{Cu} = 4.75 x 10^{-24} cm^2 derived from the MCNP calculations.

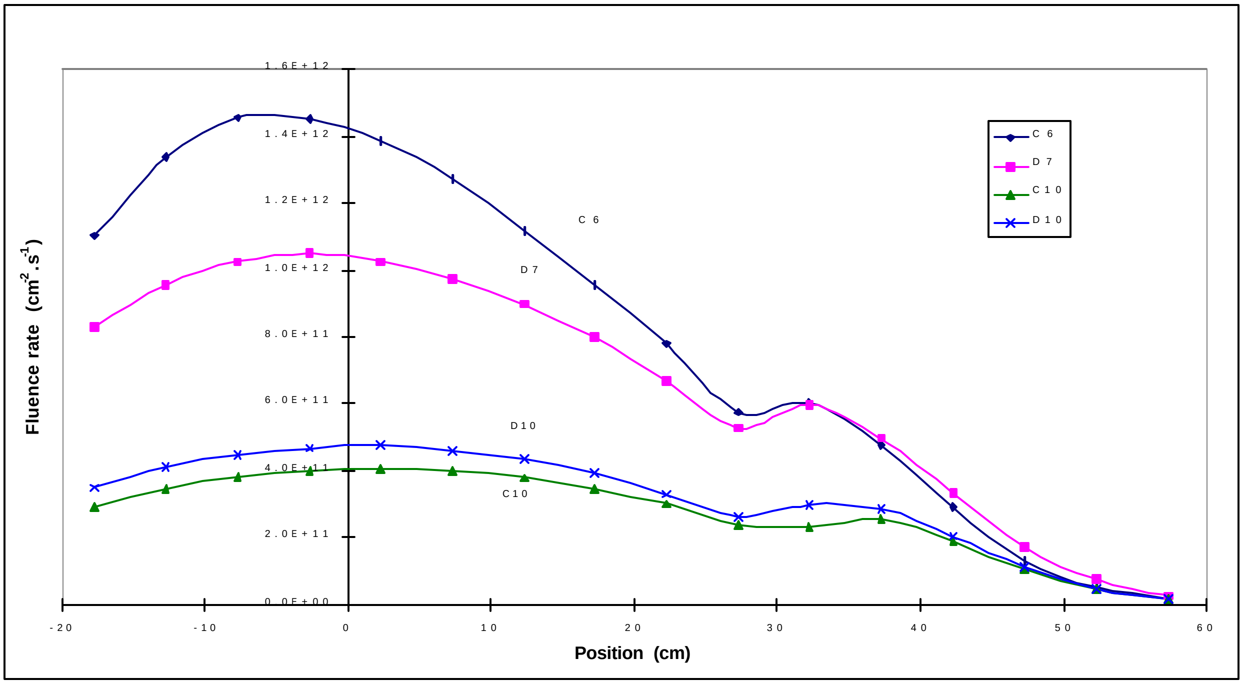

Figure 2. Thermal neutron fluence rate determined from measurement.

3.3 CALCULATION WITH NODER CODE

The power and neutron group flux densities distribution in the core are calculated with the NODER code, 3-dimensional 4 groups diffusion code developed at the NRI. The macroscopic constants for this code are prepared with the use of WIMSD4 and WIMSD4-M codes. The NODER code takes both the burn-up of the fuel units and the position of the control elements into account. Resulted thermal neutron fluence rate distributions in the centre of the fuel units along the height of C6, D7, C10 and D10 units are presented in the Fig. 3. The code also calculates the 3-dimensional power distribution in the core which is used as an input data for the MCNP and DORT calculations.

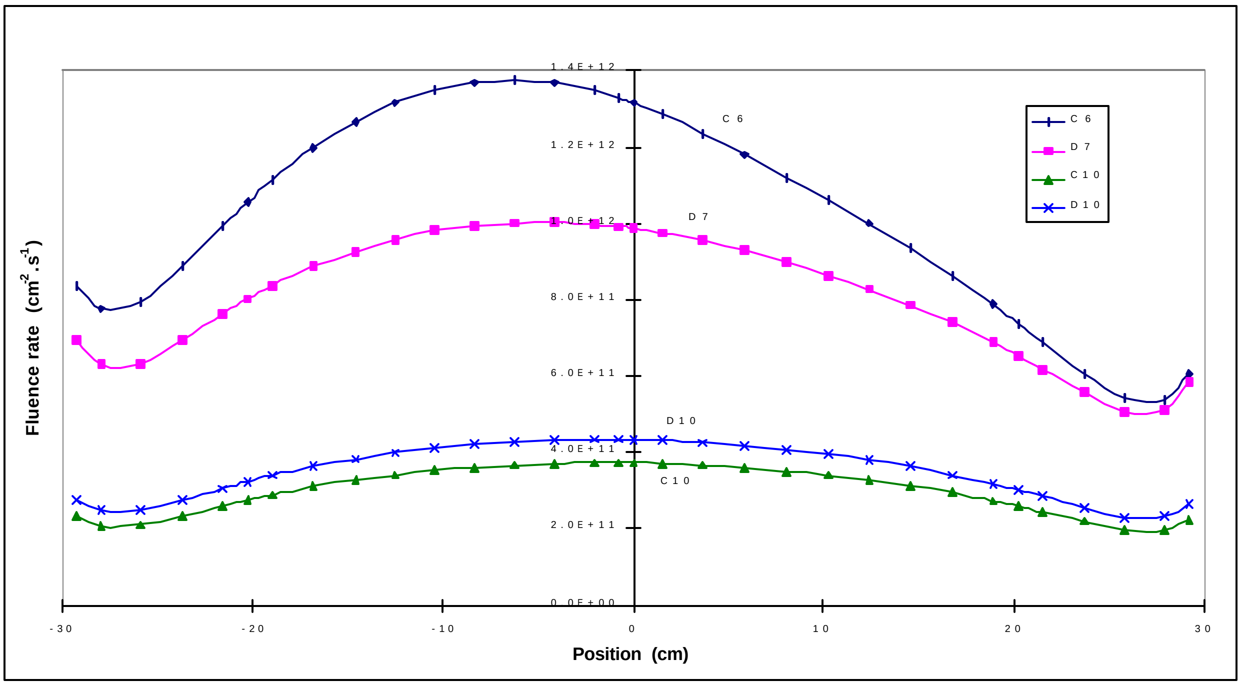

Figure 3. Thermal neutron fluence rate determined from NODER code.

3.4 CALCULATION WITH MCNP CODE

The MCNP-4B and the DLC-189 cross section library were used for the determination of the thermal neutron distribution in the desired position. The calculation was based on a detailed model of the reactor core. The model included 3-dimensional description of all the fuel, reflecting, moderating and structural units used for the core design. Also the differences in vertical position of the control rods were taken into account. The 3-dimensional neutron source distribution was transformed from the NODER code calculation. The results were normalised to the thermal reactor power which was used for the experimental verification. Resulted thermal neutron fluence rate distributions in the centre of the fuel units along the height of C6, D7, C10 and D10 units are presented in the Fig. 4. At the same time also the reaction rates (RR) of the ^{63}Cu (a,γ) reaction were calculated. From both the results an effective thermal cross section σ_{Cu} was determined according the following formula

$\sigma_{Cu} = RR/\Phi_{th} = 4.75 \times 10^{-24}\ cm^2$,

where Φ_{th} is the thermal neutron fluence rate.

The MCNP code was run on the NT 4.0 operating system on Pentium III/450 MHz computer. As the used model was quite detailed the wall clock time of the calculation was 240 hours to reach the error limit 5 %.

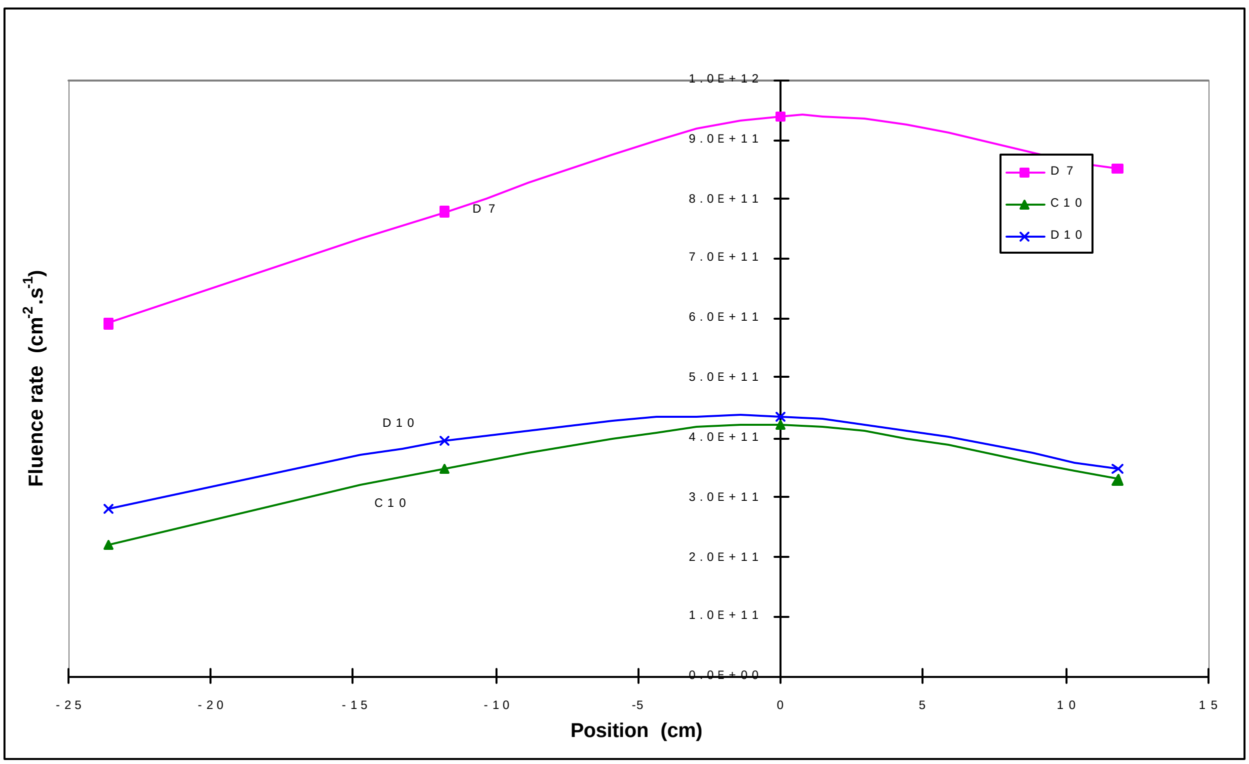

Figure 4. Thermal neutron fluence rate determined from MCNP code.

3.5 CALCULATION WITH DORT CODE

The third set of calculations was performed by the two-dimensional transport code DORT [3] with the BUGLE96 multigroup data library [4]. The neutron fission density distribution in the core was based on the NODER calculation. The three-dimensional neutron fluxes were substituted by the three-dimensional fluxes rate synthesis according to the following formula:

$$\Phi(X,Y,Z) = \Phi(X,Y)/\ \Phi(X)^{*}\ \Phi(X,Z),$$

where $\Phi(X,Y)$ and $\Phi(X,Z)$ are two-dimensional neutron flux distributions in the horizontal and vertical direction respectively. $\Phi(X)$ is one-dimensional distribution.

As the 3D flux synthesis can use only one axial source distribution in calculation, and from this respect is not the "full" tree-dimensional approach, the results are presented only for a comparison. Results of the calculation in relative units are presented in the Fig. 5.

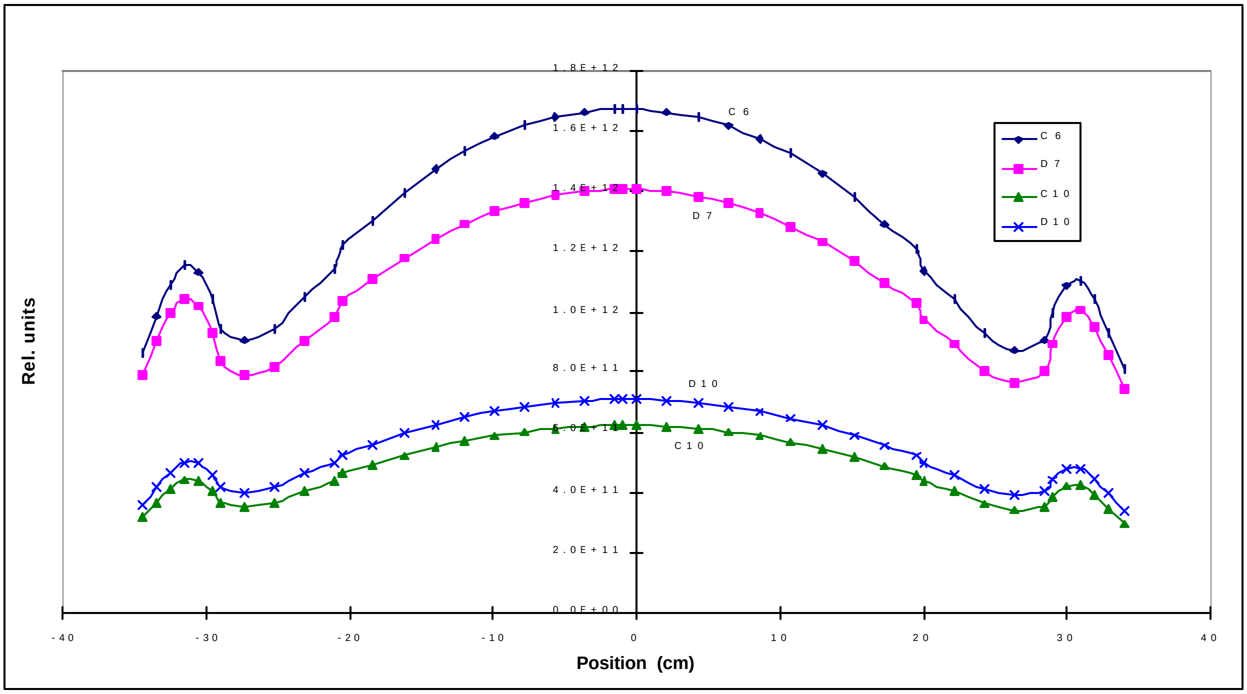

Figure 5. Thermal neutron fluence rate determined from DORT code.

3.6 COMPARISON OF THE RESULTS

The experimental results are compared with theoretical results. The thermal neutron distribution for the channel C6, D7, C10 and D10 respectively are presented in the Figures 6, 7, 8 and 9 respectively. The shape of the calculated and experimental distributions in the each channel are in a quite good agreement. The difference among the results are lower than 15%.

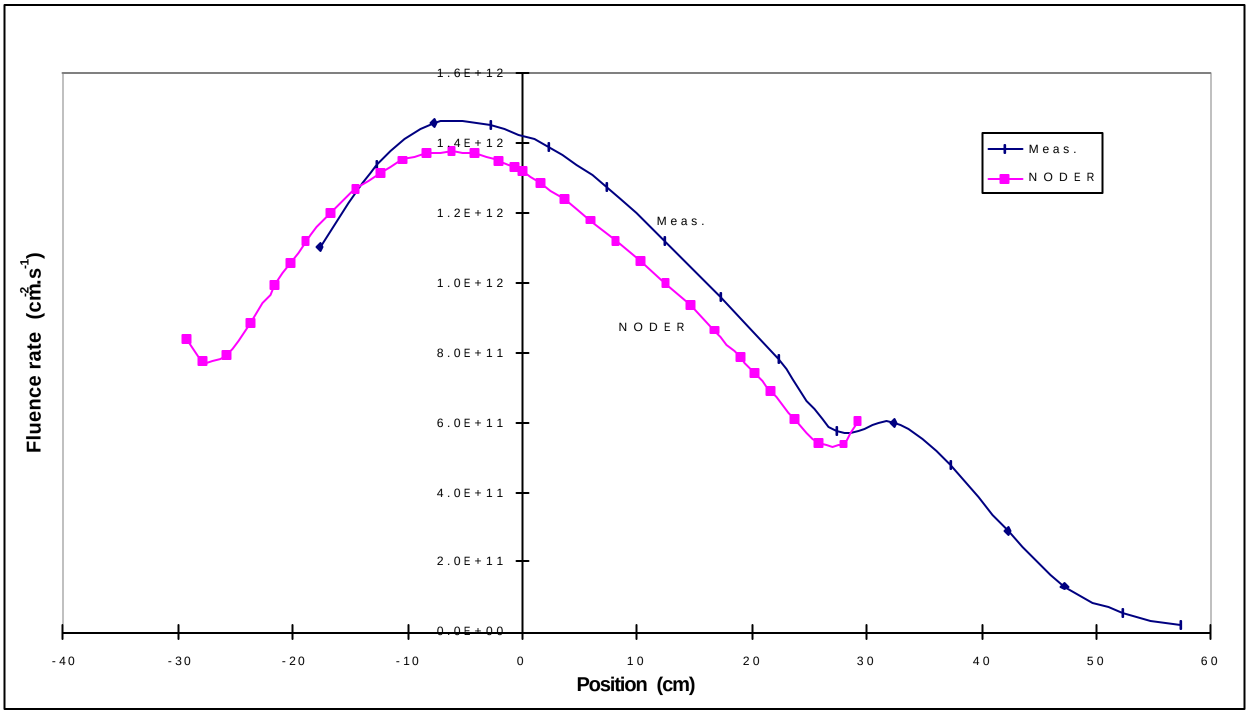

Figure 6. Thermal neutron fluence rate comparison for the C6 channel.

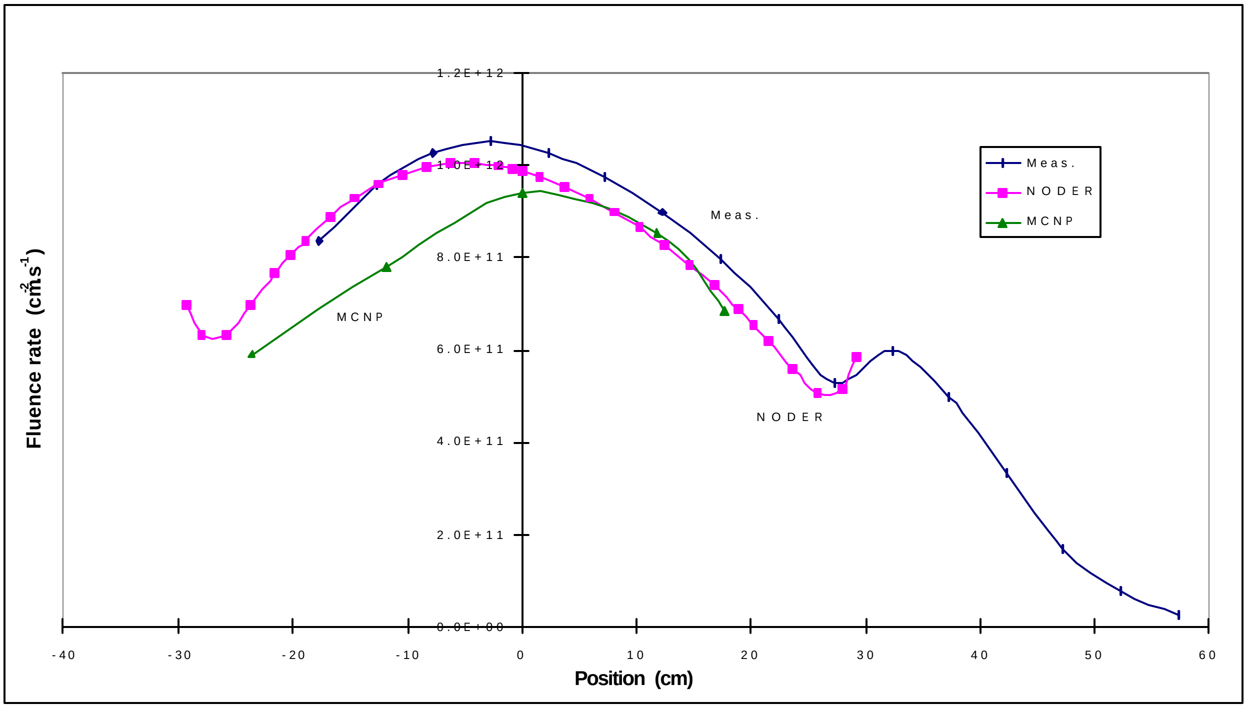

Figure 7. Thermal neutron fluence rate comparison for the D7 channel.

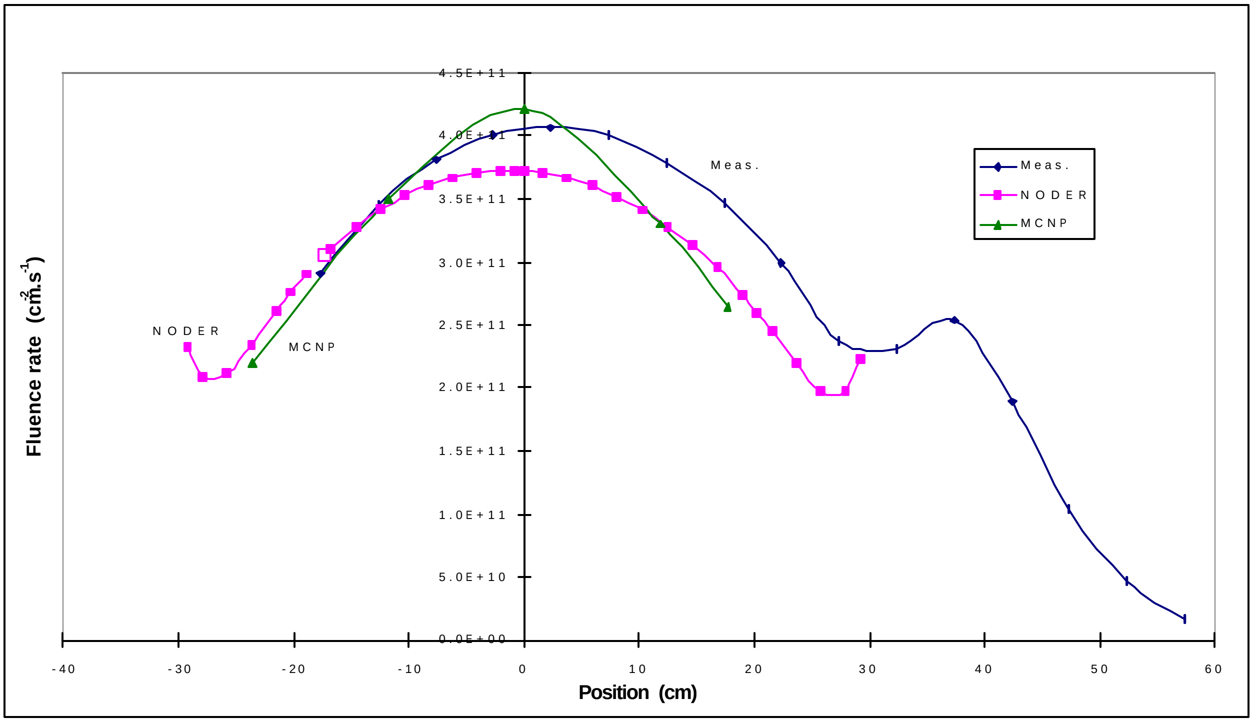

Figure 8. Thermal neutron fluence rate comparison for the C10 channel.

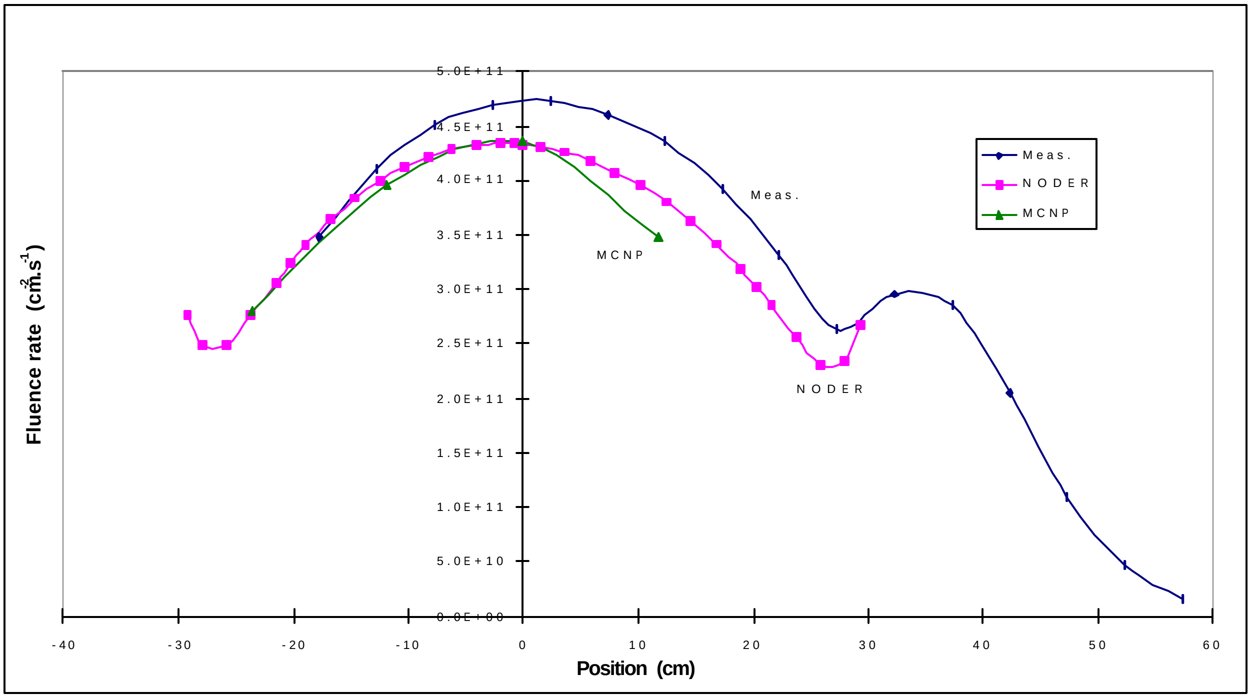

Figure 9. Thermal neutron fluence rate comparison for the D10 channel.

4. CONCLUSIONS

The presented results are used in two fields. The first one is the design of the reactor core configuration for boron neutron capture therapy experiments. The second one is the comparison of the experimental and calculated values of thermal neutron fluence rates in fuel region of the reactor. The differences are mostly less then 15 %. Especially, the agreement between the NODER calculation and the experiment is good. This means that four group 3D diffusion code is able to describe the reactor core with quite good precision. The works continue with the aim to find the sources of discrepancies and to get closer agreement between the results of the methods.

REFERENCES

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