Deterministic Analyses of YALINA-Thermal Subcritical Assembly with DRAGON-PARTISN Software

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Abstract. This study analyzes the YALINA Thermal subcritical assembly with a deterministic calculation methodology. Within this methodology, the DRAGON code, developed at Polytechnique de Montreal (Canada), has been used to generate the macroscopic cross sections for more than 50 different zones of the assembly. The DRAGON code used the WIMSD nuclear data library with 179 energy group microscopic cross sections. In the DRAGON code, the geometry of the assembly has been simplified. The generated macroscopic cross sections have been used in the PARTISN code, developed at Los Alamos National Laboratory (USA), which modeled the facility in a detailed three-dimensional geometry. The calculations focused on the kinetic parameters and the ^{3}He reaction rate profile, which has been compared with the experimental measurements.

1. Introduction

The International Atomic Energy Agency (IAEA) has a coordinated research project for accelerator driven systems intended to validate computational methods, and to characterize the performance of different subcritical systems. This coordinated research project includes the YALINA thermal experiment.[1] The YALINA thermal subcritical assembly is located in Mink (Belarus) and it can be driven by either a californium, deuterium-deuterium (D-D), or deuterium-tritium (D-T) neutron source. This work analyzes the YALINA thermal assembly with deterministic nuclear reactor codes. More precisely, the YALINA thermal assembly has been modeled by the DRAGON[2] and PARTISN codes.[3-5] The DRAGON code has been developed at the Polytechnique de Montréal (Canada) and it is used for licensing CANDU nuclear power plants in Canada. The YALINA thermal assembly is divided into 54 zones and the DRAGON code has been used to generate homogenized macroscopic cross section set for each zone. The macroscopic cross section sets generated by DRAGON have been then used in the PARTISN code which modeled the facility with 69 energy groups and 54 different zones. Some of the 54 zones are illustrated in Figure 1, which shows the PARTISN geometrical model.

In the PARTISN model three major approximations have been introduced. The first approximation ignores the cadmium and iron layers surrounding the external boundary of the graphite reflector. More precisely, the boundary of the assembly has been cut at the starting of the cadmium layer since the vacuum boundary condition is equivalent to an infinite absorber boundary condition and the cadmium cross section is large for thermal neutrons. The second approximation replaces the thin organic glass layer at the assembly boundary with a graphite layer. The thin organic glass layer is located at low importance region and both organic glass and graphite contain carbon isotopes. The third approximation fills all the reflector experimental channels with graphite. The PARTISN simulations have been performed with the S_8 pre-built angular quadrature set and with a spatial mesh containing 60, 60, and 41 intervals along the x, y, and z axes, respectively.

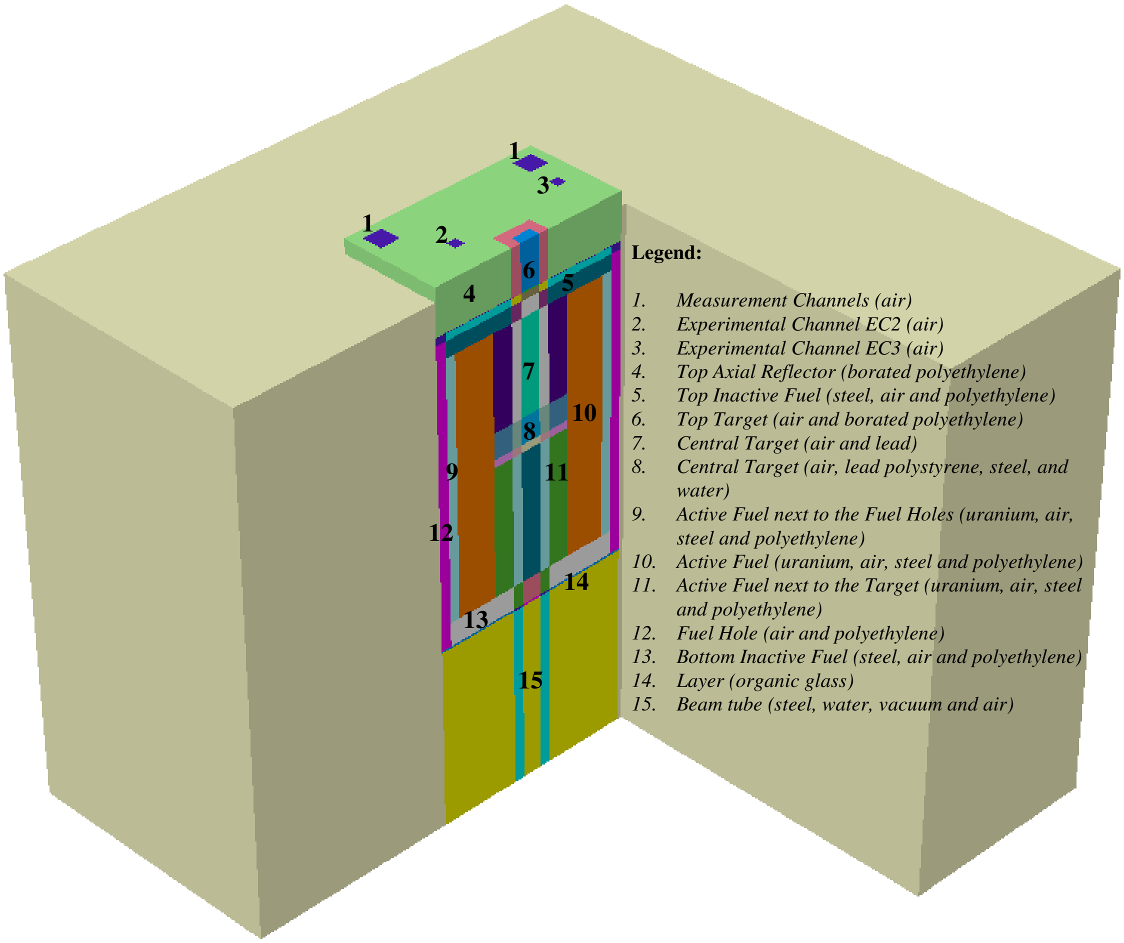

FIG. 1. Overview of the PARTISN model of the YALINA Thermal subcritical assembly.

2. DRAGON-PARTISN Software

The set of macroscopic cross sections for the PARTISN zones has been generated by the DRAGON code starting from a 172 energy group microscopic cross section library (WIMSD).[6] Three major DRAGON inputs have been used to generate the macroscopic cross sections. The first input modeled, in a two-dimensional geometry, a single EK10 fuel rod inside a 2x2 cm polyethylene box. This input generated the cross section of the fuel-moderator unit cell region. The second input modeled, two-dimensional geometrical model, 1/8 of the core including the target, fuel and reflector material zones. This input has been modified several times to generate the appropriate cross sections for the control rods holes (the control rods are always extracted during the experiments) and the experimental channels regions. In addition, this input has been also used to generate the cross sections of the graphite reflector, the measurement channels, the empty fuel holes, and the fuel close to the empty fuel holes zones. The third input modeled, three-dimensional geometrical model, 1/4 of the target zone surrounded by two concentric square rows of fuel rods. This input considered the full axial length of the facility and provided cross sections for the target, the organic glass layer (at the boundary of the inactive fuel length), the polyethylene reflector, and the aluminum extension of the fuel rods.

All the DRAGON simulations used reflected boundary conditions, an angular quadrature set with 64 angles, and the first moment of the Legendre polynomials expansion for the scattering cross sections.

In order to facilitate the coupling between DRAGON and PARTISN codes, a software interface dragon2partisn[7] has been written in C language.[8] The dragon2partisn software reads the ASCII macroscopic cross sections for a specific material from the DRAGON output and it rearranges them into the material section of the PARTISN input. The dragon2partisn software can utilize multiple DRAGON outputs files and for each of the output file the user can choose which material to process. According to the dragon2partisn material processing sequence, the macroscopic cross sections are sequentially stored in the material section of the PARTISN input. The dragon2partisn software can also process the scattering matrix up to the first moment.

3. Results

The cross sections for the fuel-moderator unit cell calculated by DRAGON have been compared with those ones calculated by MCNP/MCNPX[9-10] in section 3.1. The kinetic parameters, neutron flux profile and neutron spectra obtained by PARTISN have been calculated with those ones obtained by MCNP/MCNPX in sections 3.2, 3.3 and 3.4, respectively. The ^{3}He (n,p) reaction rate calculated by PARTISN has been compared with the experimental measurements in section 3.5.

3.1 Fuel-Moderator Unit Cell Cross Sections

Figure 2 illustrates the total, absorption, fission macroscopic cross sections and the $\nu\Sigma_f$ (number of secondary neutrons multiplied by the fission cross section) of the homogenized fuel-moderator unit cell region of the YALINA thermal assembly. The unit cell region is an infinite long 2x2 cm polyethylene square cell with reflecting boundary conditions containing a single EK10 fuel rod.[1] DRAGON results show an excellent agreement with the MCNP/MCNPX results.

3.2 Kinetic Parameters

Table I summarizes the multiplication factor, the delayed neutron fraction and the prompt neutron lifetime for three different fuel loading configurations: 216, 245, 280. The three configurations have different number of fuel rods as explained in more details in Reference 1. An increase in the number of EK10 fuel rods augments the reactivity of the assembly. The results obtained by PARTISN for the multiplication factor differ less than 150 pcm from the results obtained by MCNP/MCNPX. The delayed neutron fraction and the prompt neutron lifetime have been calculated by PARTISN. The adjoint neutron flux has been calculated either in critical or in source mode; both modes produce similar results. The results are in good agreement with those obtained from Monte Carlo simulations.

When the delayed neutron fraction and the prompt neutron lifetime are not weighted on the adjoint flux (indicated by adjoint=1 in Table I), the results considerably differ.

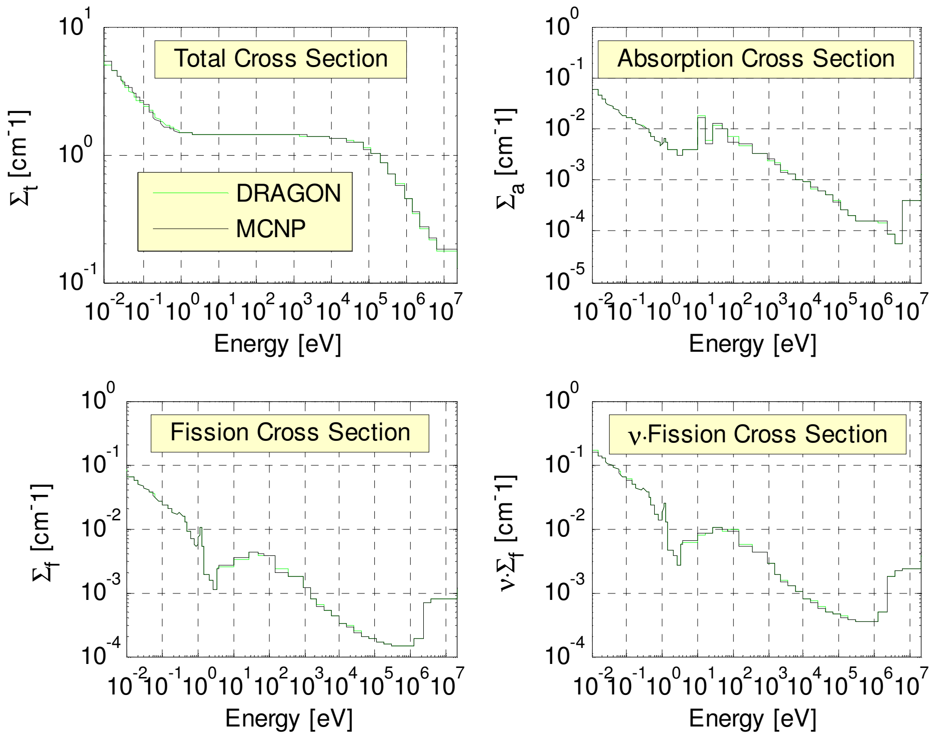

FIG. 2. Homogenized macroscopic cross sections calculated by DRAGON and MCNP.

TABLE I: Kinetic parameters of YALINA Thermal

		Configuration		
Code	**Library**	**216**	**245**	**280**
		Multiplication Factor k_{eff}		
MCNPX2.6f	ENDFB-6.6	0.87902± 9	0.91911± 4	0.95833± 7
PARTISN4	ENDFB-6.8	0.88059	0.91962	0.95857
		Delayed Neutron Fraction β_{eff} [pcm]		
MCNPX2.6f	ENDFB-6.6	769±12	783± 6	774±11
PARTISN4	JEF-2.2 adjoint=1	673	673	673
PARTISN4	JEF-2.2 critical adjoint	786	776	768
PARTISN4	JEF-2.2 source adjoint	784	775	767
		Prompt Neutron Lifetime l_p [μs]		
MCNPX2.6f	ENDFB-6.6 adjoint=1	260	255	242
PARTISN4	JEF-2.2 adjoint=1	271	255	231
MCNPX2.6f	ENDFB-6.6	74±2	80±2	82±2
PARTISN4	JEF-2.2 critical adjoint	81	80	82
PARTISN4	JEF-2.2 source adjoint	81	79	82

3.3 Neutron Flux

The neutron flux profile calculated in the experimental channels EC1 and EC5 and set by the fission neutron source is shown in Figures 3 and 4. These calculations refer to the fuel configuration with 216 EK10 fuel rods and the fission neutron source. The profile follows a cosine shape along the active fuel length which spreads over 50 cm. Once again the PARTISN results have been compared with the MCNP results showing a very good agreement.

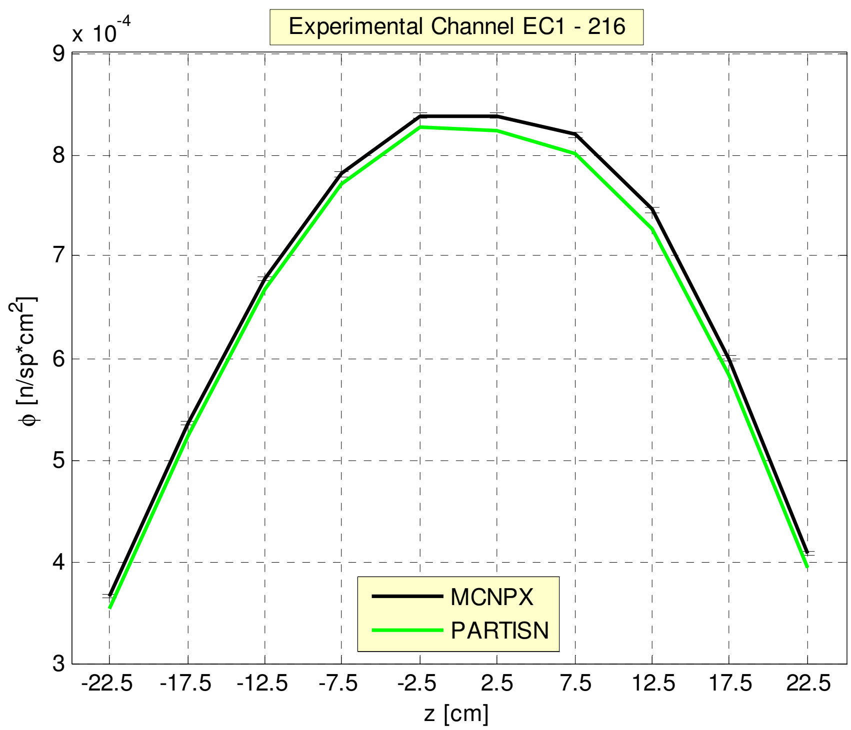

FIG. 3. Neutron flux profile in EC1 calculated by PARTISN and MCNP.

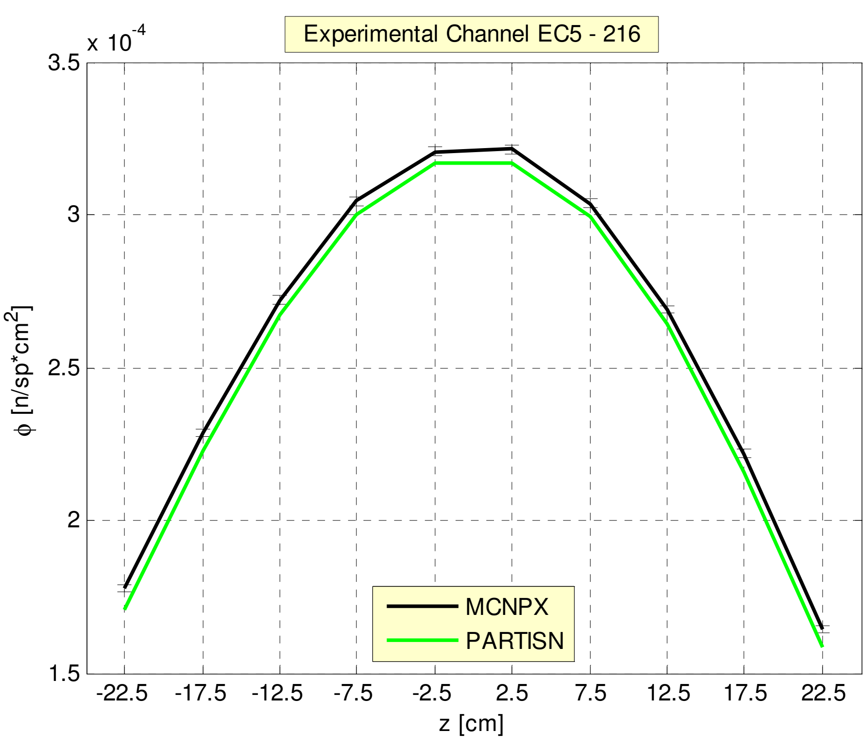

FIG. 4. Neutron flux profile in EC5 calculated by PARTISN and MCNP.

3.4 Neutron Spectrum

Figures 5 and 6 depict the neutron spectra at the center of EC1 and EC5 experimental channels for the same fuel configuration analyzed in the previous section. These spectra are obtained from criticality calculations. The spectrum calculated in EC5 is much more thermalized than the spectrum in EC1. The EC5 experimental channel is located in the moderator region of the facility while EC1 is located next to the target at the center of the assembly. In the slowing down energy region, the shape of the neutron spectrum calculated by PARTISN shows some dips and peaks which have no physical meaning. These irregularities are due to the DRAGON collapsed cross sections and they disappear if 172 energy group cross sections are used. However, these irregularities are still under investigation.

3.5 ^{3}He (n,p) Reaction Rate

The ^{3}He reaction rate calculated by PARTISN has been compared with the experimental measurements in Figure 7. In this case the assembly is driven by a californium neutron source, which is placed in the middle of the active fuel length and the beam tube is removed. In the plot, the experimental results are normalized so the maximum experimental value at the z=0 matches the PARTISN value since the absolute source neutron source intensity is not measured. The agreement is very good except for a small deviation in the negative direction of the axial coordinate. This behavior has been also observed in the MCNPX results and is still under investigation.

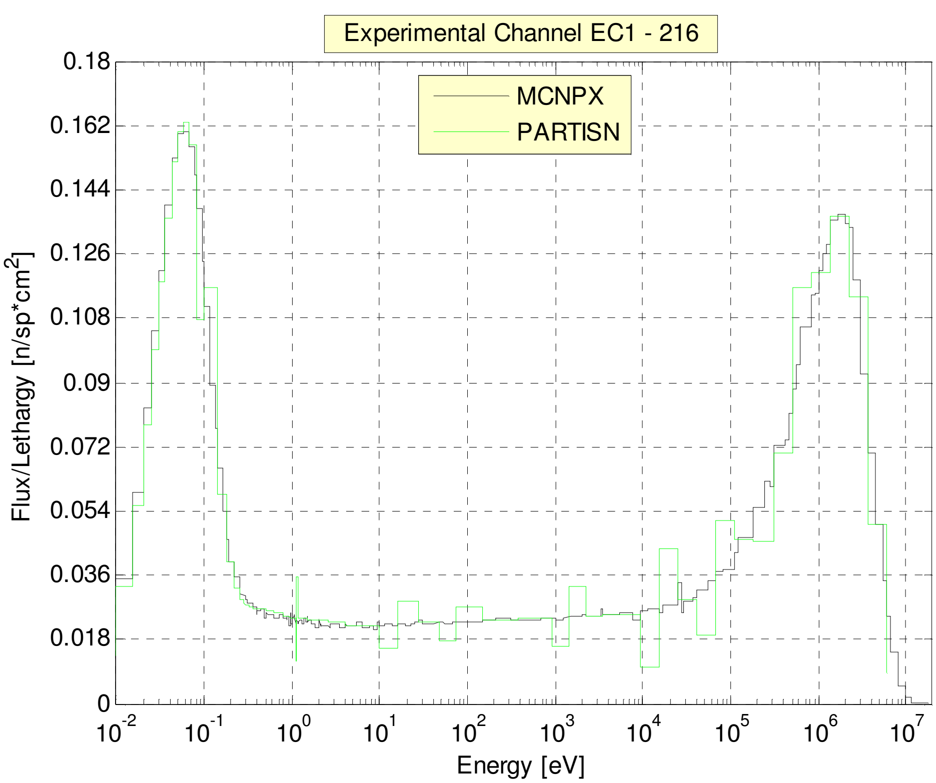

FIG. 5. Neutron spectrum in EC1 calculated by PARTISN and MCNP.

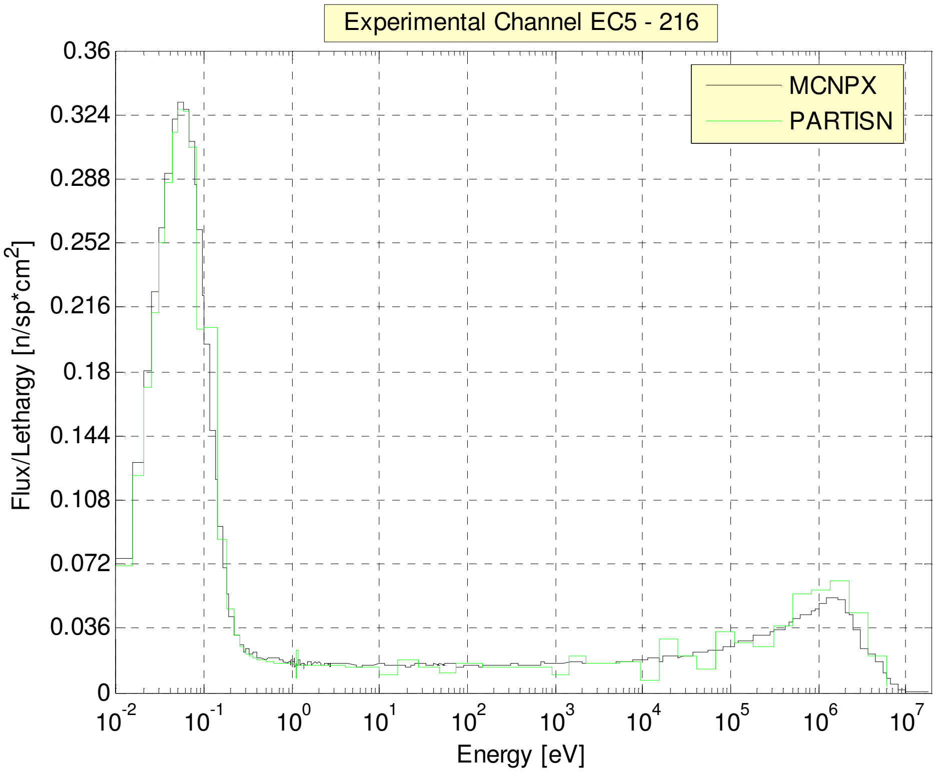

FIG. 6. Neutron spectrum in EC5 calculated by PARTISN and MCNP.

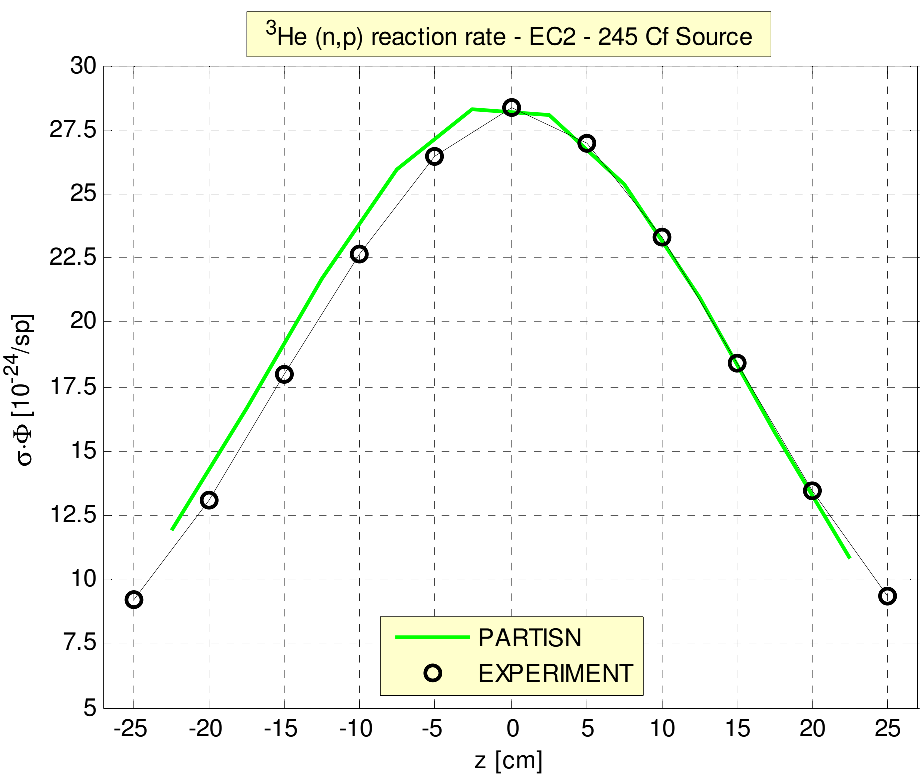

FIG.7. 3He (n,p) reaction rate in the active fuel lengh. Californium neutron source.

4. Conclusions

This study used DRAGON and PARTISN computer codes to analyze the YALINA thermal assembly performance. The results of the deterministic methodology agree very well with those obtained by Monte Carlo codes (MCNP/MCNPX). The comparison between deterministic and Monte Carlo results covered the kinetic parameters (multiplication factor, delayed neutron fraction and prompt neutron lifetime) the neutron flux profile, and the neutron spectra in the experimental channels of the subcritical assembly. Some irregularities in the neutron spectrum have been observed and they are related to the DRAGON cross sections, however, the irregularities disappear if 172 energy groups cross section set is used. The ^{3}He (n,p) reaction rate calculated by PARTISN has been compared with the experimental measurements and the comparison shows a good agreement.

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5. References

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