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Modeling of Existing Beam-port Facility at PSU Breazeale Reactor by using MCNP5

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The Radiation Science and Engineering Center facilities at the Pennsylvania State University (PSU) include the Penn State Breazeale Reactor, gamma irradiation facilities, and various radiation detection and measurement laboratories. Due to inherited design issues with the current arrangement of beam ports and reactor core-moderator assembly, the development of innovative experimental facilities utilizing neutron beams is extremely limited. Therefore, a new core-moderator location in PSBR pool and beam port geometry was needed to be developed. A study is underway with the support of DOE-INIE funds to examine the existing beam ports for neutron output and to investigate new moderator and beam-port designs to produce more useful neutron beams.

The overall system for this study consists of two major parts, the core model and beam port model. Core calculations are performed by using a three dimensional nodal diffusion code ADMARC-H. Beam port calculations are performed with the MCNP code. An interface program has been developed at PSU to link the diffusion code to the neutron transport code. The MCNP model consists of the D_2O tank, graphite reflector block, and beam port tube with their surroundings. The results of the PSU package show good agreement with the experimental data.

KEYWORDS: **MCNP, Penn State, beam port design, TRIGA Reactor, Core Design, Energy spectrum.**

1. Introduction

The Radiation Science and Engineering Center (RSEC) facilities at The Pennsylvania State University (PSU) include the Penn State Breazeale Reactor (PSBR), gamma irradiation facilities, and various radiation detection and measurement laboratories. The mission of the RSEC, in partnership with faculty, staff, students, alumni, government, and industry members, is to safely utilize nuclear technology to benefit society through education, research, and service. The PSBR is the nation's longest continuously operating reactor that went critical in 1955. The PSBR is a 1 MW, TRIGA with moveable core in a large pool and with pulsing capabilities. The core is located in a pool of de-mineralized water. When the reactor core is placed next to a D_2O tank and graphite reflector assembly near the beam port (BP) locations, thermal neutron beams become available for neutron transmission and neutron radiography measurement from two of the seven existing beam ports.

When the PSBR reactor was built, MTR type fuel elements with active length of 24” were used. With the MTR fuel the beam port arrangement did not limit the maximum neutron output. However, in 1965, the original 200 KW reactor core and the control system was replaced by advanced General Atomics

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TRIGA core with an active fuel length of 15” and analog control system. The new core is capable of operation at a steady-state power level of 1000 KW with pulsing capabilities up to 2000 MW for short (milliseconds) period of time. In 1991, the reactor console system was upgraded to an AECL/Gamma-Metrics dual digital/analog control system.

With TRIGA fuel, only one beam port is at the centerline of the core active area, four beam ports are five inches below the centerline and two are eleven inches below the centerline (below the active fuel region). The PSBR beam port layout is shown in Figure 1. Only two beam ports are currently being used (beam ports shown as red in Figure 1). BP #4, which is located axially at the centerline of the reactor core is used for research, primarily neutron radiography and radioscopy, and BP #7 with its lower neutron flux level is used for service activities involving neutron transmission measurements. Due to inherited design issues with the current arrangement of beam ports and reactor core-moderator assembly, the development of innovative experimental facilities utilizing neutron beams is extremely limited. Therefore, a new core-moderator location in PSBR pool and beam port geometry was needed to be developed. A study is underway with the support of DOE-INIE funds to examine the existing beam ports for neutron output and to investigate new moderator and beam-port designs to produce more useful neutron beams.

2. Description of Work

The overall system for this study consists of two major parts, the core model and beam port model. Core calculations are performed by using the diffusion code ADMARC-H [1], which utilizes a few-group cross-section library developed with HELIOS [2]. Since the geometrical and material configuration of the beam-port facility is complex the MCNP [3] code is used instead of the deterministic codes because of its geometrical flexibility. An interface program has been developed at PSU to link the diffusion code to the neutron transport code. This interface basically reads the ADMARC-H output then computes the source term for MCNP and finally prepares the necessary MCNP input card at the requested format. The schematic view of the overall simulation and the system are given in Figures 2 and 3, respectively.

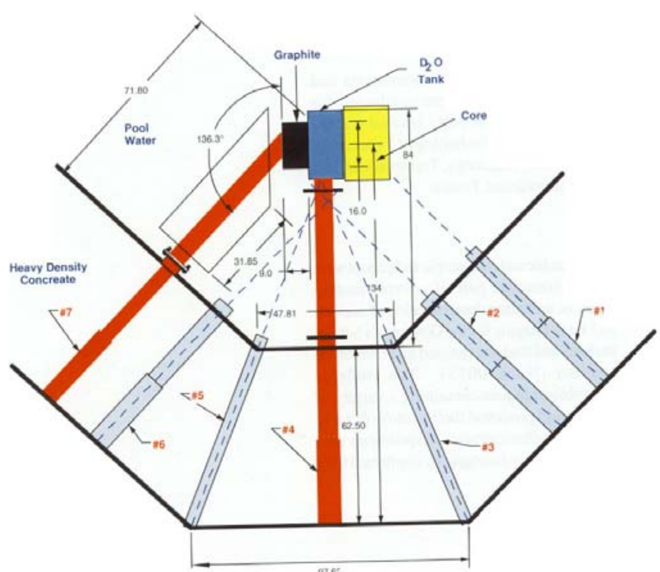

Fig.1. PSBR beam port layout with D_20 tank

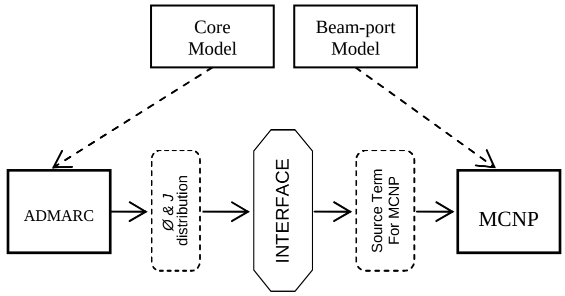

Fig.2 The schematic view of the simulation package

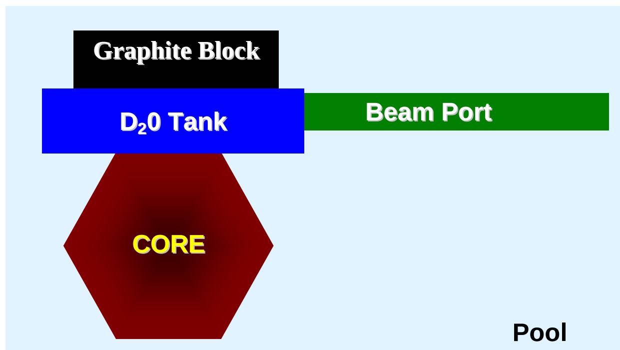

Fig.3 The schematic view of the system

2.1 The Core Model

The PBSR TRIGA core was first loaded in 1965 with 8.5^{wt}% ZrHx-U fuel. Since July 1972, the core has been patially reloaded with fresh 12^{wt}% ZrHx-U fuel elements, six at each reload. Currently, the PBSR is operated using core cycle 51 as shown in Figure 4. The uniform lattice in PBSR forms a hexagonal shape. The center of the core is the location of the central thimble (the water hole), which is surrounded by hexagonal rings of fuel. There are 94 fuel rods; 30 of them are 12 ^{wt}% and 64 of them are 8.5 ^{wt}%, both with a 20 ^{wt}% uranium-235 enrichment. Three control rods (shim, regulating and safety) are fuel-follower control rods driven by motor. They are composed of graphite at the top and bottom, fuel and absorber (borated graphite) in the middle. The fourth control rod is the transient rod, the only control rod without fuel material, which is driven by an electro-pneumatic during steady-state operation.

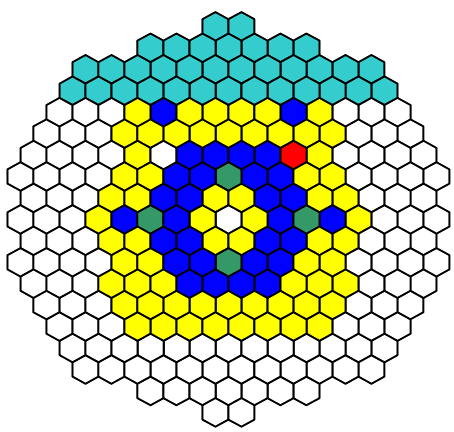

Fig.4. The ADMARC-H Core model

The dark blue and yellow cells in Figure 4 represent the fuel assemblies with 12.0^{wt}% and 8.5^{wt}% uranium, respectively. The green cells show the locations of the control rods and the red cell represents the location of the external source which is in the reactor. The white cells and cyan cells are the light water and heavy water reflectors, respectively. This core loading is used in the core calculations and it has been validated using measured data from TRIGA [1].

2.2 The Interface Module

Core calculations and the MCNP model is linked with an interface developed at Penn State. The source for the MCNP model is taken from the core calculations with ADMARC-H, which is capable of producing cell averaged fluxes or incoming partial currents at each surface of the cells. Since we need the incoming source into the D_2O tank from the reactor for this study, in ADMARC-H, the incoming partial flux output option is used. An example of ADMARC-H cell and the numbering of the surfaces on this cell is shown in Figure 5. At the D_2O tank-reactor interface, the incoming partial currents for surfaces 5 and 6 of each D_2O tank cell are taken from the ADMARC-H output to define the source term for that node. Then these incoming partial currents are added using vector addition rules to compute the source for a given cell.

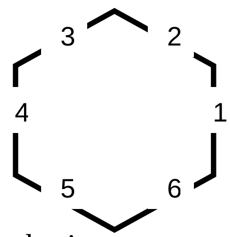

Fig.5. An ADMARC-H Cell and surface numbering

A schematic view of the source computation for an ADMARC-H cell is shown in Figure 6.

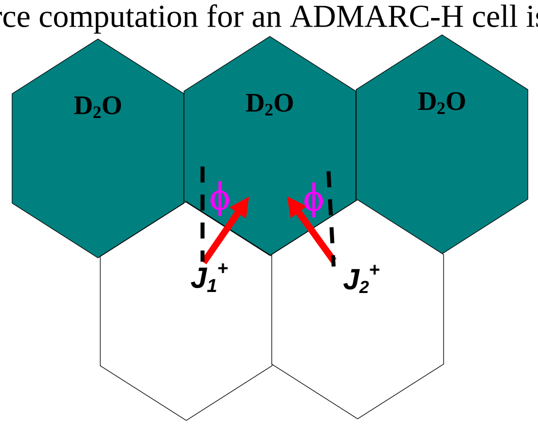

Fig.6. Schematic view of the source computation for an ADMARC-H cell

The cell source at the D_2O tank-reactor core interface is calculated at each node for each group as shown in Equation 1

$$Source(i) = J_1(i)A_1 \cos\phi + J_2(i)A_2 \cos\phi \tag{1}$$

where, *i* represents the energy dependency, *i=1,2.*

The ADMARC-H is able to take into account only two energy groups, fast and thermal. Therefore, the output is also in two energy groups. In order to perform calculations with a more detailed energy spectrum, we first performed a preliminary lattice cell calculation for an average reactor cell and analyzed the energy spectrum out of this average cell and used this result to re-compute the energy dependence of the source term for the MCNP model. The total flux is calculated as

$$TotalSource = \sum_{i=1}^{2} Source(i) \tag{2}$$

Then this total source is redistributed over the given energy range using the calculated energy spectrum for an average fuel cell shown in Figure 7. Finally, the interface program prepares the input card for the source term in the MCNP5 source term input format

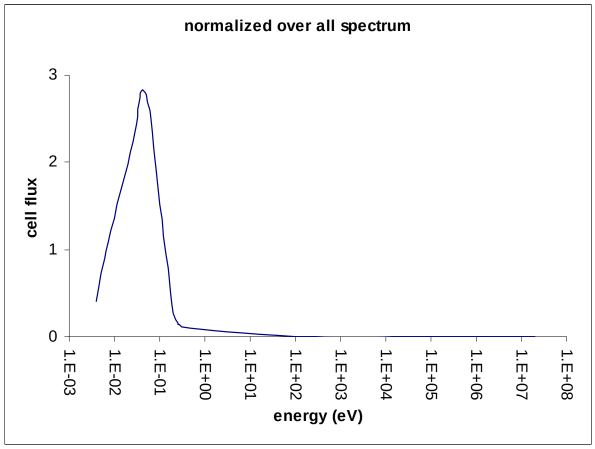

Fig.7 Energy spectrum used to redistribute the total source term.

2.3 The MCNP Model

The MCNP model consists of the D_2O tank, graphite reflector block, and beam port tube with their surroundings. The source term for the MCNP model is taken from the ADMARC-H calculations using the interface program. Preliminary runs showed that modeling of the whole system with MCNP is computationally expensive. Therefore, the overall MCNP model is broken up into two parts, namely a D_2O tank model and a beam-port model. The MCNP part of the simulation is performed in two steps. First, the D_2O tank model, which contains the source information from the interface code and geometry data for the D_2O tank is run. Then, the output of the D_2O tank model is used as the input for the beam-port model. Finally the beam-port model simulation is performed with this input data.

2.3.1. D_2O Tank Model

The D_2O tank model includes the D_2O tank next to the reactor and the graphite reflector adjoint to the D_2O tank. The top and side views and the dimensions of the D_2O tank model are given in Figures 8 and 9, respectively.

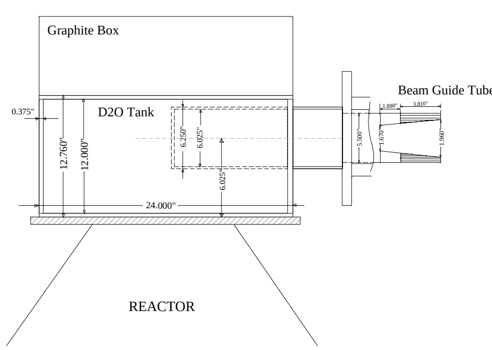

Fig.8 The top view of the D_2O tank model

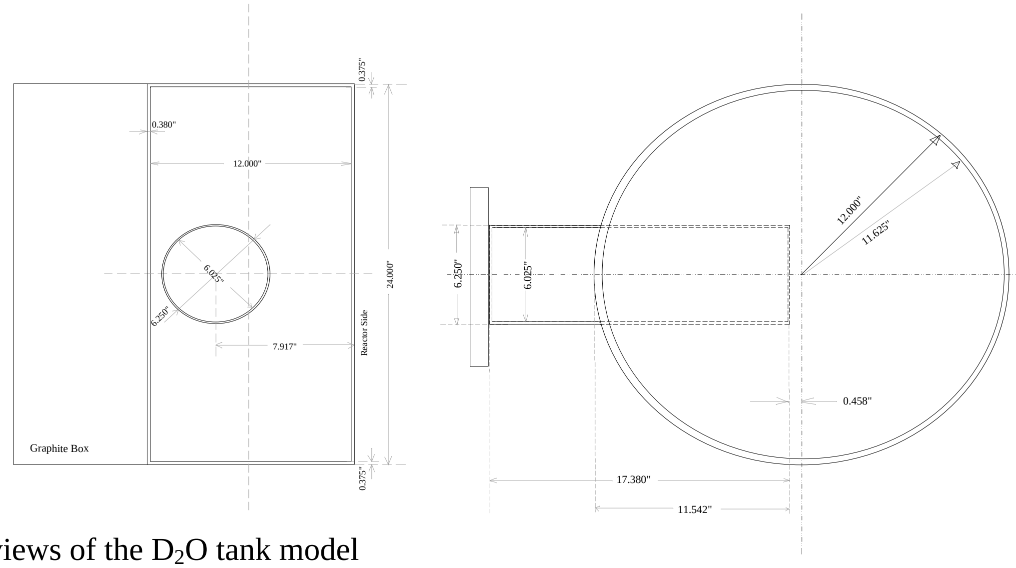

Fig 9 Side views of the D_2O tank model

The D_2O tank serves as the link in between the beam tube and the reactor. It gets the source from the core calculations and performs the MCNP calculations and provides the source term for the beam tube model. The result of the D_2O tank calculations is shown in Figure 10.

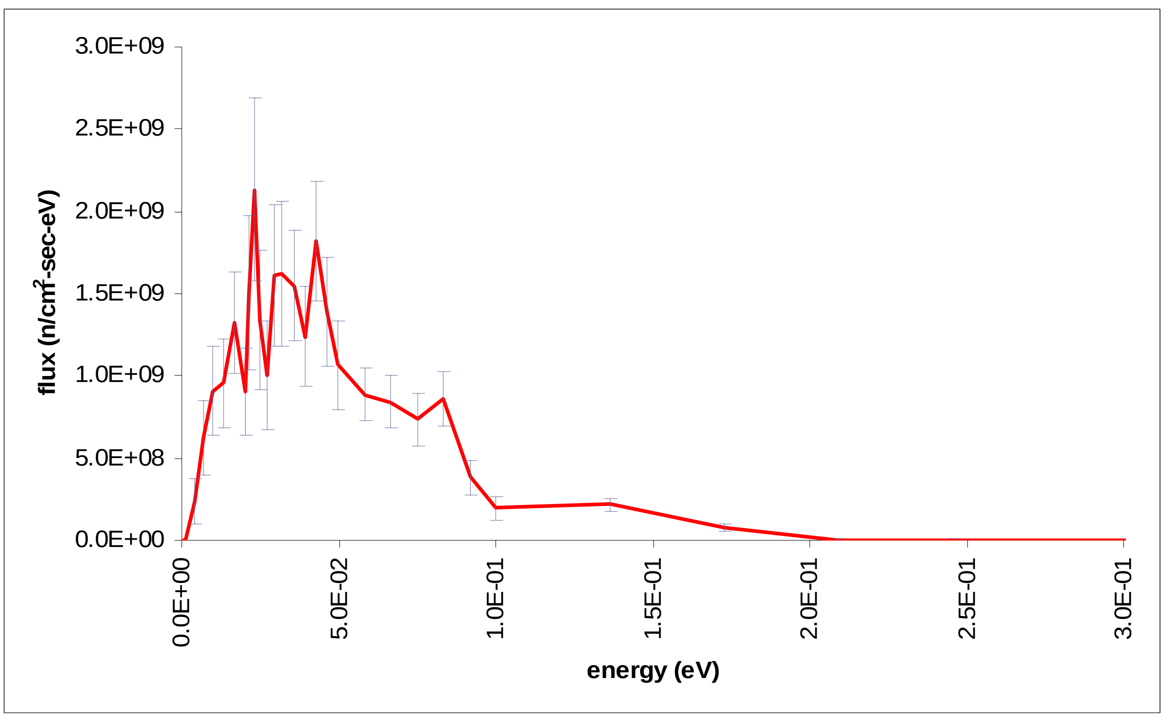

Fig.10 Comparison of D2O tank model results with the experimental data

2.3.2. Beam Port Model

The beam port is located at the axial centerline of the reactor core. Since the available experimental data was taken at the exit of BP #4 a model was developed using the physical dimensions of the port and the borated aluminum aperture and the lead filters and concrete collimators. The top view and the dimensions of the model, and the insertions are shown in Figure 11.

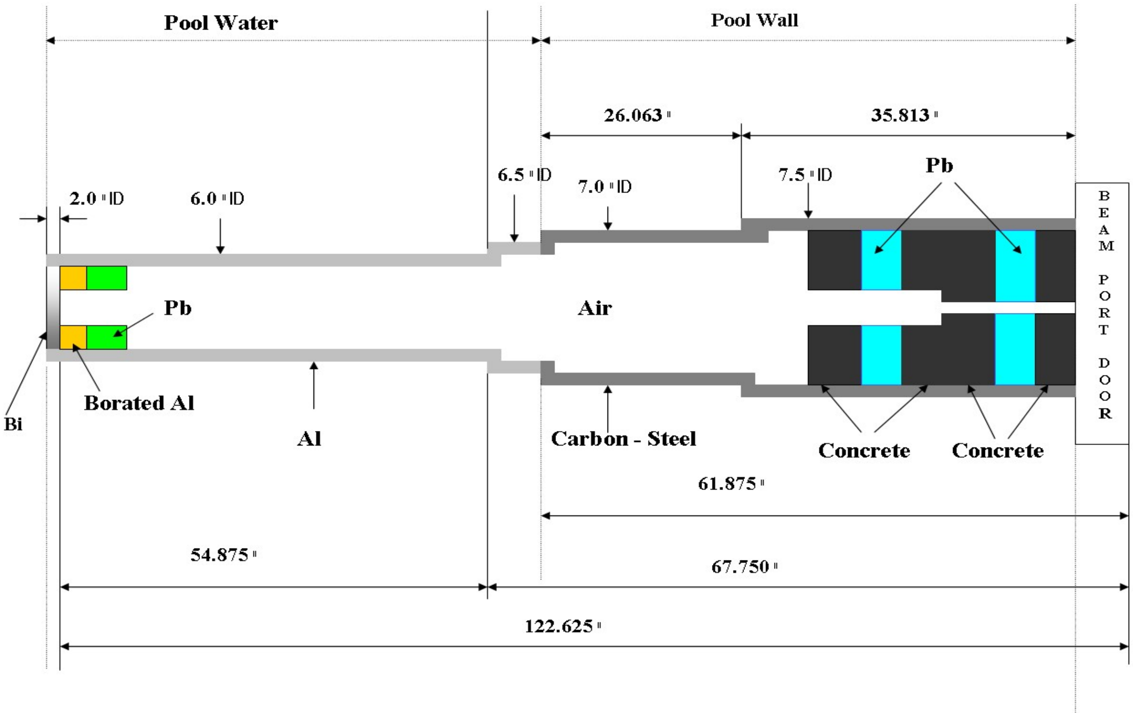

Fig.11 The dimensions and the schematic view of BP #4

The BP #4 configuration was modeled using MCNP. The output of the D_2O tank model is used as the incoming source into the BP #4 model. The results of the MCNP model were compared with the experimental data. A plot of this comparison is given in Fig.12.

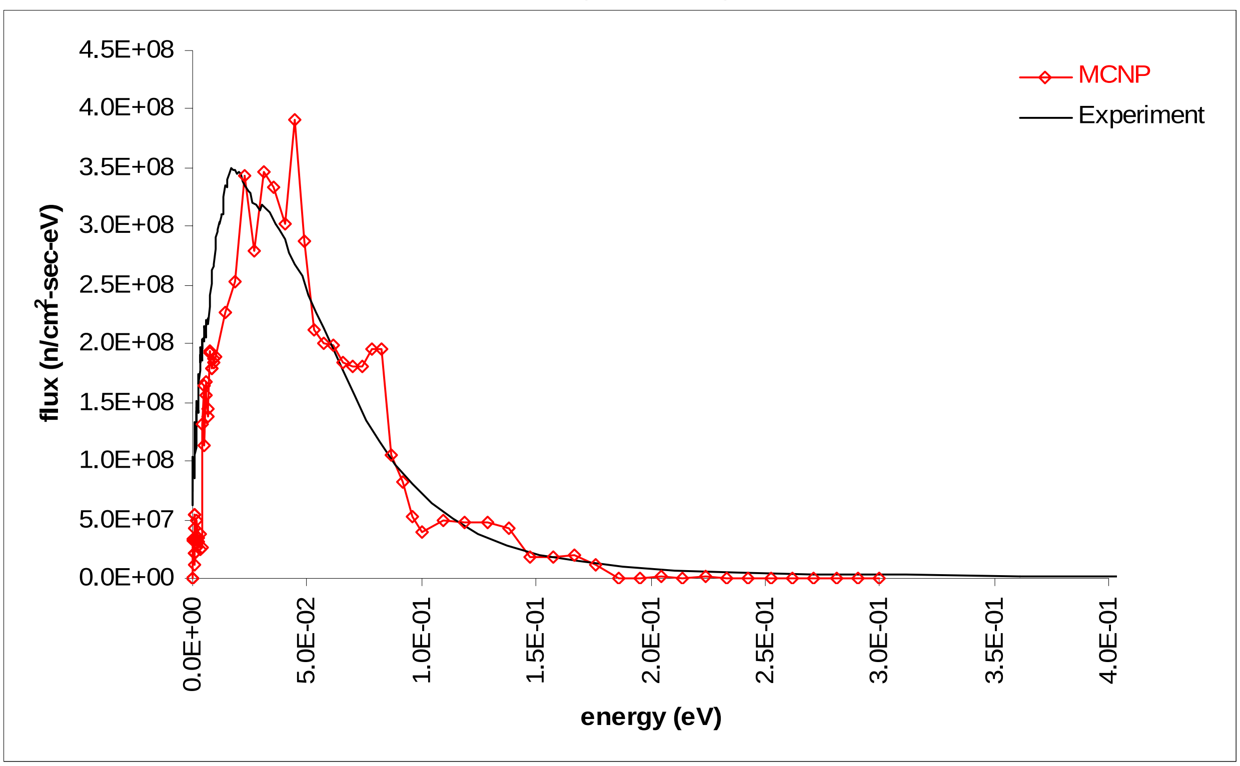

Fig.12 Comparison of BP #4 model result with the experimental data

3. Results and Discussions

The comparison of the PSU Computational Package with the experimental data, Figure 12, shows that the computer models for the combined core and beam-port calculations give good agreement with the experimental data. However, the error band for the D_2O tank calculations is high as shown in Figure 12. Therefore, we need to continue to perform more histories in the MCNP calculations for the D_2O tank model.

4. Work Underway and Future Work

4.1 Work Underway

The results show fairly good agreement with the experimental data. In order to further test the accuracy of the two-step MCNP model, a full MCNP model that includes both the D_2O tank and the beam port models is also being performed.

The modeling of the beam ports #5 and #7, which are shown in Figure 1, were also performed by using the same methodology and the computational tools. The calculations for these beam ports are also being performed to determine the neutron flux at the end of each of these beam ports.

A study for the optimization of the beam port entry hole location for BP #4 is also underway. The main goal of this optimization study is to find the best location of the beam port entry position in the D_2O tank in terms of the maximum thermal flux.

4.2 Future Work

Once the tools and methodologies are verified and the models are finalized, the next step is to design a better reflector-beam port configuration. The main steps of this study include a new design for the D_2O tank, new design for the graphite box and different beam port designs.

The final step of this study will be to include the core model into the MCNP calculations. Once we finish developing the necessary tools for the core model, we will include the core model into the MCNP model and combine this model with the D_2O tank and beam port models to eliminate all the intermediate steps.

References

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