USING MCNP IN THE DESIGN OF NEUTRON SOURCES AND NEUTRON BEAMS

Daniel F. Hergenreder, Carlos A. Lecot and Osvaldo P. Lovotti

INVAP S.E., Nuclear Projects Department, Nuclear Engineering Division
F.P. Moreno 1089
(R8400AMU) S.C. de Bariloche, Río Negro, Argentina

ABSTRACT

The calculation methodology used to design Cold, Thermal and Hot Neutron Sources and their associated Neutron Beam Transport Systems is presented. The design goal is to evaluate the performance of the neutron sources, their beam tubes and neutron guides at specific experimental locations in the reactor hall as well as in the neutron hall.

The Monte Carlo method is a unique and powerful tool to transport neutrons. Its use in a bootstrap scheme appears to be an appropriate solution for this type of system. The proper use of MCNP as the main tool leads to a fast and reliable method to perform calculations in a relatively short time with low statistical errors.

Keywords: MCNP, neutron, source, guide, transport.

I. INTRODUCTION

During the design of neutron sources and their associated neutron beams and neutron guides it was observed that some changes in the design variables produce an increment in the neutron flux at the source or guide entrance, but produce a decrease in the neutron flux at the end of the guide at specific experimental locations in the reactor hall as well as in the neutron hall.

It is necessary during the design stage to use a powerful tool to transport the neutrons as of their creation in the core conditions. The increase in the calculation capacity of new computers makes it possible to use the Monte Carlo technique during the design stage.

The Monte Carlo method used in a bootstrap scheme appears to be an appropriate solution for this type of system. The MCNP, ref. [1], is the tool used to create the neutrons in the core conditions, moderate the neutrons in the moderator or reflector and neutron sources and to transport them from the neutron source to the guide entrance through the beam tube.

From the guide entrance to the experimental location the neutrons are transported through the neutron guide (typically a super mirror). The transport in the neutron guides is carried out using the Monte Carlo method taking into account the reflective properties of the neutron guide as a function of neutron energy, and a full description of the geometrical conditions such as dimensions, curvature radii and non alignment effects.

II. SCOPE

Consider a thermal source constituted by heavy water placed near the core. The thermal source has two associated beam tubes located in opposite directions. The region between the beams constitutes the main portion of the thermal source. During the analysis of the optimized distance between beam tips, the thermal flux was analyzed at the thermal source center, at the neutron guide entrance and at the specific experimental location (end of neutron guide).

Fig. 1 shows the thermal flux increment at the center of the thermal neutron source when distance between beam tips is modified, but the “useful” neutron flux at the guide entrance or at the end of guide behaves differently.

Figure 1. Thermal Flux During the Thermal Source Design.
Fig. 1 shows that the thermal flux at the guide entrance and the thermal flux at the end of guide behave similarly because the design parameter does not change the neutron spectrum at the guide entrance.

However, there are situations where the design parameter modifies the neutron spectrum at the guide entrance and, therefore, the neutron guide transmission.

This situation is found frequently in the design of cold neutron sources, because most of the design parameters modify the capacity of neutron moderation of the source and modify the cold neutron spectrum.

For design purposes, the guide transmission is defined as the ratio between the neutron flux at the end of guide and the neutron flux at guide entrance.

Fig. 2 shows how the guide transmission changes when a cold neutron source parameter, such as the CNS diameter, is modified.

A high number of neutrons are created by the KCODE source and only those that arrive at the region involved are recorded in the surface source.

The technique of using a surface source is adequate while there are no modifications in operation or geometrical conditions outside the region involved, especially in the path from the core to this region. In the region involved it is possible to make changes, except in the region near the boundary of the surface source.

The design is carried out using the surface source as a neutron source and the DXTRAN technique is applied to increase the number of particles that arrive at the guide entrance. The particles that arrive at the guide entrance are registered in a file (ptrack file) using the PTRAC card.

This is when the bootstrap technique starts. The ptrack file is filtered in order to reduce the information about each particle. The information that needs to be preserved is the position at which the particle enters the guide (x, y, z), the direction of the particle (u, v, w), its energy and its weight. During the filtering stage it is possible to select an energy or weight window as filter variable.

Once filtered, a new coordinate system is used for the particles recorded in the ptrack file, where the x axis is the neutron guide direction, the y axis is the direction of the neutron guide height and the z axis is the direction of the neutron guide width.

Now, the particles recorded in the ptrack file are ready to be transported through the neutron guide.

Transport through the neutron guides is carried out using the Monte Carlo implicit transport method. It means the particles that have been scattered in the plates of the neutron guide, with an angle for which the reflectivity of the plate is greater than zero, will arrive at the end of the guide. The weight of these particles will be reduced according to the reflectivity for each collision.

The Monte Carlo analog transport method is also available, but it requires a higher number of particles to obtain the same statistical errors.

The neutron guide transport program allows us to take into account the reflective properties of the neutron guide as a function of neutron energy, as each plate of the neutron guide could have different reflective properties. The geometrical flexibility admits a full description of the geometrical conditions such as dimensions, curvature radii and random non-alignment effects.

It is possible to analyze the particle distribution at different positions of the neutron guide and to make an energy, angle or weight spectrum. At the end of the neutron guide, another ptrack file is created with the particles that arrive at the end of guide, that would permit another neutron guide transport if it were necessary.

The main hardware requirements are the calculation and data storage capacity. Typically, the size of a surface source is 1 GB and the size of a filtered ptrack file is 500 MB.

**III. DESCRIPTION OF THE CALCULATION LINE**

The MCNP is the main tool of the calculation line. With MCNP it is possible to make a full description of the operation conditions and geometrical characteristics in three dimensions.

The MCNP model includes full details of the core: typical burnup distribution (radial and axial), burnable poisons, enrichment distribution and critical control rod positions. The full detail description extends to the surroundings of the core where the irradiation facilities, neutron sources and neutron beams are located. Typically this region has Beryllium or Heavy Water as reflecting material and is contained by the concrete reactor block or the light water tank.

Criticality calculations are used by the MCNP to generate neutrons from fission in the critical core. A surface source is created to include the interest region, for example, the cold neutron source and its associated beam tubes, or the thermal source and its associated beam tubes.
IV. RESULTS

In order to show how this technique works we analyzed a particular problem. A cold neutron source (CNS) placed near the core is calculated. Associated to the CNS there is a beam tube that has a neutron guide inside. The CNS is cylindrical in shape. In order to increase the cold neutron flux that leaves the CNS in the beam tube direction, a cavity is designed.

A prism with an elliptical section is the shape proposed for the cavity. The large diameter of the ellipse is equal to the width of the beam tube, it is perpendicular to the beam tube direction and tangent to the CNS cell. The small diameter of the ellipse is the design variable.

A large value of the small diameter makes it possible for the neutrons coming from the proximity of the CNS center to enter the beam tube. The neutrons coming from the CNS center are cooler neutrons and the neutron guide will transport them more efficiently. On the other hand, a large value of the small diameter increases the volume of the cavity and reduces the volume of the CNS cell, i.e., reduces the CNS capacity to moderate neutrons.

A value equal to zero for the small diameter of the ellipse means that the cavity is not utilized.

We must find the value for the small diameter of the ellipse that produces the maximum cold neutron flux at the experimental location, i.e., the optimum value.

Fig. 3 shows the MCNP model for the CNS region where it is possible to see the CNS cavity, the beam tube and the CNS cell.

Fig. 4 shows the cold neutron flux (0 to 10 meV) at the guide entrance.

![Figure 4. Cold Neutron Flux at Guide Entrance.](image)

Fig. 5 shows the cold neutron flux at the experimental location after a transportation length of 3 meters in the neutron guide.

![Figure 5. Cold Neutron Flux at Experimental Location.](image)

The guide transmission as a function of the small diameter value was shown previously, in Fig. 2, (see section II).

Fig. 6 shows the gain of the cold neutrons as a function of their wavelength when the cold neutron is placed in the reflector. The gain is obtained at the experimental location placed 3 meters from the guide entrance.
V. CONCLUSIONS

This calculation methodology where the MCNP is used as the main tool to transport fission neutrons, allows a full description of the main operation conditions of the reactor and geometrical characteristics inside and outside the core. The further transport through the neutron guides using the Monte Carlo technique allows us to continue the calculation with a full description of the system conditions.

The results obtained at the experimental locations have low statistical errors and permit us to determine the behavior of the neutron flux as a function of each design variable.

Due to the capacity of the calculation methodology to make a full description of the system, it is possible to analyze, simultaneously, the effects of the design variables on different parameters such as the core reactivity and the neutron fluxes required by the facilities placed around the core.

REFERENCES