MCNP Report
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1 Result

1.1 Exercise 1 Simple Neutron Source in a Bucket of Water

1.1.1 Input file

Energy cut-off is not applied so that very slow (thermalized) neutrons to interact and let further reactions take place.
Therefore the lower bound of the 1st bin is 0.
The temperature of the cross-section data in databse 42(ENDL92, acquired by Lawrence Livermore National Laboratory) were acrquired at T=300K, as shown in 91-99, and is subsequently adjusted down to 20.4 C^o (2.53×10^{-8}MeV). Either way, thermal effects should significantly affect that falls into the first 3 energy bins. (0-10^{-9}MeV, 10^{-9}-10^{-8}MeV,10^{-8}-10^{-7}MeV respectively). The energy group structure was not chosen to be finer because of
Insert the pictures of the cross sections of the geometry here, caption with cell number and material number

1.1.2 Output file

By examining the first 50 particles(using `PRINT 110`), the source was confirmed to be a point souce 2cm above the centre of the bottom of the tank's internal surface; and the majority of the particles have initial energy $E < 4$ MeV as expected when they are distributed according to the Watt spectrum for neutron generated by ^{235}U+n(thermal)

Insert Watt Spectrum .png

plot variation of VoV, fom, , as nps increases to 20000

1. These results are not reliable because the statistical checks (
   insert number of them not passed
) are not passed, meaning that some reactions are not sampled enough for us to be confidence about the frequencies of their occurance.
   Additionally, some bins in the PDF are empty (seen table 161 of tally 22) suggesting that insufficient number of neutrons is sampled to approximate a continuous distribution.

2. The total fluences Φ are simplay calculated by formula $NA\sum_i(\Psi_i)$
   where each Ψ_i refers to the flux calculated for the ith energy bin *per history*.
   A = Effective area for surface flux tallying, where the particle passing through still had non-zero weight.
   N = number of histories
   Giving *create table

3. The neutron spectra
   (insert logx histograms using pandas, [dividing by the difference in upper and lower class

boundaries so that it gives number density instead?].)

Copy and modify table above to show their respective means

The average standard flux ϕ i.e. number of neutrons passing through each unit surface area of interest, and the implication (fewer gets their because of the inverse square law)

The hardness of neutron spectrum decreases as follows: hardness at base (Tally 32)>long side (Tally 12)>short side (Tally 22), because distance of them increases in that order. More neutrons can penetrate the base at the energy its creation energy without being moderated by the water; therefore the first bin of tally 32is more filled than that of tally 12.

The ratio of neutron flux in the slow/Cadmium region (<0.01 MeV) to neutron flux in the resonance regions (>0.01MeV) builds up since the latter gets slowed down/captured by the resonance peaks much faster than the rate of consumption of the former, as they travel through the material.

Note that the 1st bin (0 to 1×10^{-9} MeV) is below the thermal energy of room temperature; neutrons of these energies are strongly affected by the free gas thermal treatment of the interaction cross sections; additionally due to the large cross section in the thermal region which increases as the neutron slows, very few neutrons of these energies can travel far enough to be counted by these surface flux tallies. Therefore this bin always have the largest relative errors among all bins.

To ensure that no bins have relative error larger than 0.10 to ensure that the results are reasonable, enough histories must be ran to allow this energy bin in all 3 tallies to be filled sufficiently, in addition to passing all 10 statistical checks.

The tally probability distributions (histograms shown in table 161's) all show reasonably Gaussian distributions (and the cumulative probability distributions all show reasonably sigmoidal distributions) without the majority of the counts falling into a particular bin; therefore there is no inherent reason for using a finer spacing; but for the purpose of visualizing the spectrum with a higher resolution, a finer group structure with trifolded bin density (still logarithmically spaced, 3 bins per decade) was used.

Even when the spacing of energy groups is trisected in this manner, the 1st bin (0 to 1×10^{-9} MeV) is still the bin that receives the least counts. In a real problem this would've been merged with the 2nd bin to form a single, larger bin to tally up all the completely thermalized neutrons, increasing its count so that it converges to a precise enough value in a shorter CPU time; but according to the constraint of the problem, there must be one bin with 10^{-9} MeV as the upper limit. Therefore the only option left is to increase the number of histories such that the results of this bin converges.

Even after running 500000 histories, the relative error for this bin remains high (0.2599, 0.4236, 0.3681, for tally 12, 22, and 32, respectively) despite the fact that all other statistical checks were passed and the fom has alreay converged to a stable asymptotic limit.

The number of histories to run is chosen as *, empirically shown to give a result that converges (passes all 10 statistical checks) in a reasonable CPU time of * minutes.

1.2 Exercise 3: Criticality

Easier geometry to define if we use a sphere. However, using a square box + bottom allows a surface flux detector to be applied such that the The weight was subsequently reduced to 0.1 such that the simulation runs faster by ignoring

The neutron mean free path is found on a Cell particle activity table (using `PRINT 126`) giving average track mpf (cm) in cell λ =cm in a rough simulation with fewer number of cycles. The concrete floor is then cut down to extend in each direction by 3λ.

Additionally, the neutrons contribution to k_{eff} after passing from the tank to the floor was

examined, and was determined to be insignificant.

Using SDEF's default option, a "generic Watt spectrum"(
Quote MCNP manaul) of neutron was created.

The neutron source was defined using the position = distribution 1 (`POS=D1`) option in the `SDEF`
card, where distribution are the centres of the cylinders.

This allowed for very fast convergence of the results, where the k_{eff} settled down in less than
10 generations; but to be safe 100 generations were skipped (because it isn't computationally
expensive to do so).

1.2.1   Questions

- Examine and report upon the estimate of k_{eff} with cycle number given in the output.

- Are you confident in the final reported result and its uncertainty? Justify your answer.

    - make sure that the 10 statistical checks pass on all cells

- In addition to Monte Carlo stochastic uncertainties what other uncertainties may need
  to be considered in a criticality safety assessment?