The Monte Carlo Method: Versatility Unbounded in a Dynamic Computing World
Chattanooga, Tennessee, April 17-21, 2005, on CD-ROM, American Nuclear Society, LaGrange Park, IL (2005)

FAST MONTE CARLO DOSE CALCULATIONS FOR ALL PARTICLES: ORANGE

Steven C. van der Marck and Alfred Hogenbirk
Nuclear Reasearch and Consultancy Group NRG
P.O. Box 25, NL-1755 ZG Petten, the Netherlands
vandermarck@nrg-nl.com; hogenbirk@nrg-nl.com

ABSTRACT

We present dose calculations with the ORANGE code, which is based on MCNP(X). ORANGE can speed up dose calculations with MCNP by many orders of magnitude for photon and electron beams. This work has now been extended to neutrons, protons, and all particle types that are transported by MCNPX. We demonstrate the accuracy of the dose calculation and the speed of the calculation by comparison with EGS for a photon beam, and by comparison with experiment for a high energy neutron beam. We conclude that Orange does an accurate and fast Monte Carlo dose calculation for all particles.

Key Words: Monte Carlo; dose calculation; proton; neutron; photon

1 INTRODUCTION

When one wants to plan a radiotherapy treatment, one has to rely on a good dose calculation. The accuracy of dose calculations is an area of some concern in conventional radiotherapy, for external photon and electrom beams [1]. This has led to the imminent introduction of Monte Carlo dose calculations in treatment planning software for external photon and electrom beams. For neutron capture therapy, Monte Carlo dose calculations for treatment planning was used already earlier, because of the difficulties in calculating neutron dose by other means. However, in this case, the Monte Carlo program, a speeded-up version of MCNP, was used without electron transport, and kerma factors were used in combination with a track length estimator to calculate the flux [2, 3, 4]. For yet another treatment modality, proton beams, the dose calculation is especially difficult. The typical energy of incident protons is high, $100-200$ MeV, the nuclear cross section data are not always known or know accurately, and the localization of the dose around the Bragg peak poses a challenge.

In this paper we present a dose calculation method that can handle all the above situations and is fast at it too: ORANGE.

2 ORANGE, BASED ON MCNPX-2.5E

Earlier we reported on the development of Orange for neutron dose [5] and for electron and photon dose [6, 7]. Orange was based on MCNP-4C3 [8] with two main additions: an extra tally F9 for the dose (based on energy balance per interaction) and a grid definition that is independent from the material geometry. Now we have done the same based on mcnpX-2.5e [9], which in turn is based on a.o. MCNP-4C3, but can transport many more particles. The additional feature we had to put in is that our F9 dose tally needs to work properly when particles are transported by the built-in physics models that are part of mcnpX. These models are used whenever there are no nuclear cross section data available.

We want to emphasize that the method of energy balance per interaction is both the most accurate possible and the most versatile. For photon and electron Monte Carlo programs, such as EGS [10] and Penelope [11], it is the method of choice because nothing less will suffice to reach the accuracy required. For a neutron dose, one needs to incorporate the so-called Q-value of certain interactions, such as (n,α) or fission. This is easy to do with our method, but not for e.g. the *F8 tally of MCNP(X), which uses energy balance per cell (not per interaction). Alternatively one could incorporate these Q-values in kerma factors, such as in NCTplan [2], but then one cannot account for the influence of electron transport. Finally, when using energy balance per interaction, the deposited dose is by definition consistent with the transport simulation. It is therefore easy to switch to different nuclear cross section data, or to different physics models for particle transport. In all cases the dose calculation will be based on the transport data used, without the need to first calculate e.g. new kerma factors. In the case of protons this may prove rather useful.

Having extended Orange to mcnpX-2.5e, we have a dose calculation that is applicable to almost any situation imaginable, involving electrons, photons, neutron, protons, α-particles, ... The only exclusion would be heavy ion therapy, since e.g. C-12 nuclei are not transported by mcnpX.

3 RESULTS

We demonstrate the quality of Orange by 2 examples. The first one is a computational benchmark for external photon beams, the ICCR-XIII benchmark [13]. It consists of an 18 MV photon beam impinging on a slab phantom, with 3 cm water, then 2 cm aluminum, 7 cm lung, and 18 cm water. The results of the Orange F9 tally are plotted in Fig. 1, together with those of the mcnpX Mesh Tally 3, and of EGS4. It is clear from the Figure that the Orange F9 results are close to those of EGS4 (within 2%). On the other hand, those of mcnpX Mesh Tally 3 are too low. Moreover, for this phantom with $61 \times 79 \times 150$ voxels, mcnpX Mesh Tally 3 was $1000\times$ (!) slower than Orange F9.

Our second example is an experiment performed at LANL recently [12], where a high energy neutron beam was directed at a $30 \times 30 \times 30$ cm^3 phantom with tissue equivalent liquid. The incident neutron energies ranged up to 750 MeV, with a peak around 100 MeV. The dose was measured in the phantom at central positions, roughly 1 cm apart. The source strength was also measured, which enables the calculation to be normalized at that source strength. In other words, there is no room for fitting any variable.

The results of Orange for this experiment are shown in Fig. 2, together with those of mcnpX Mesh Tally 3, and with the experimental values. Again we see that the results of Mesh Tally 3 are lower than for Orange. The Orange results are clearly closer to the experimental values. The last two experimental points (at $d = 28$ and 29 cm) are outliers, see also the discussion in Ref. [12]. The only deviation is just after the beam has entered the phantom, where the experimental points

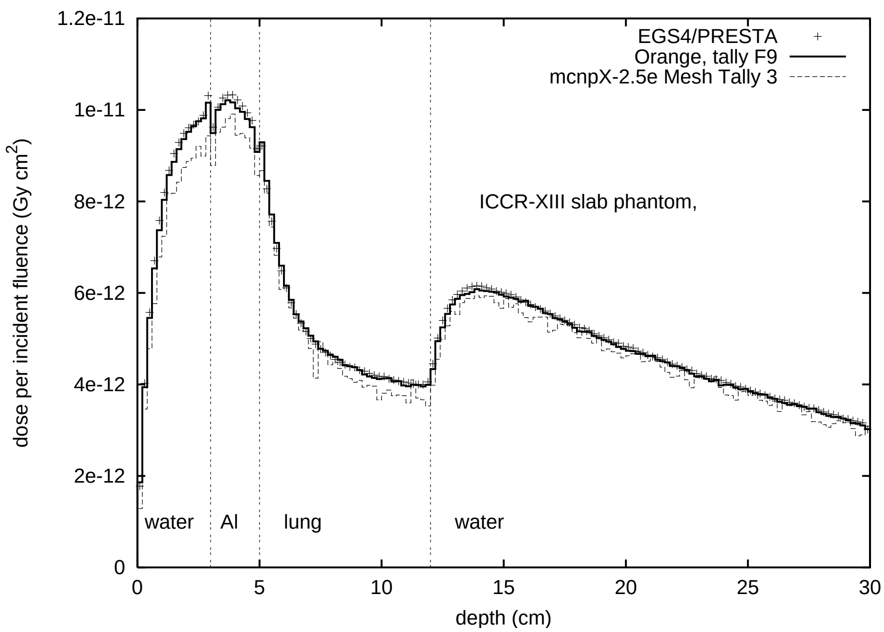

Figure 1: Results for the dose per incident fluence for the ICCR-XIII benchmark phantom, calculated by Orange, mcnpX, and EGS4.

at $d = 1, 1.5$, and 3 cm are higher than the calculated ones, for both Orange and mcnpX.

It is interesting to decompose the dose into contributions from various particle types. In Fig. 3 we plot these dose contributions, showing that both neutrons and protons give sizable dose depositions in this case. Therefore this example may serve as a first step to prove that the proton dose is well accounted for in Orange.

Also for this example, Orange is faster than mcnpX. The relative difference is smaller, because the transport simulation of neutrons takes more time than for photons. Nevertheless, the F9 tally uses 16% extra time (for 30^3 voxels) on top of the time needed for particle transport alone, compared to an additional 129% for Mesh Tally 3. When the number of voxels is increased to 60^3, F9 adds another 5%, and Mesh Tally 3 adds another 680%.

In both examples, the Orange F9 results are close to the reference results (EGS4 and experiment respectively), whereas the mcnpX Mesh Tally 3 is too low. It should be said, though, that the LANL experiment has been performed for several filtered beams too, and that for those the mcnpX results are sometimes closer to the experimental values than the Orange ones. The only difference with the experiment shown in Fig. 2 is in the beam characterisation, so the reason for these different findings is most probably to be found there. In all cases, however, Orange results are higher than those of mcnpX, even though they are based on exactly the same particle tracks.

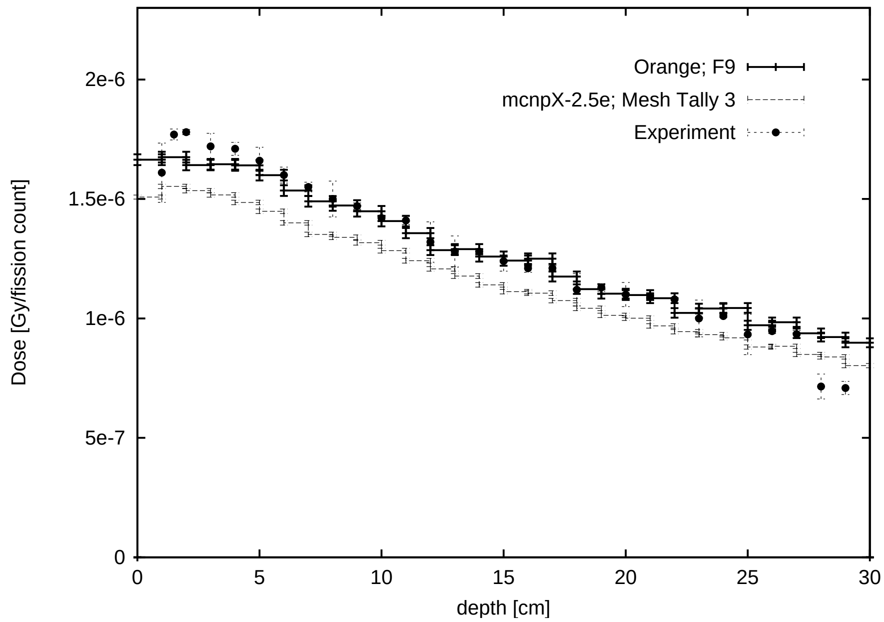

Figure 2: Results for the dose per fission count for the high energy neutron beam experiment, as calculated by Orange, and mcnpX.

4 CONCLUSIONS

We have developed Orange, a dose calculation method based on mcnpX-2.5e. It can calculate the dose for all particles that are transported by mcnpX-2.5e, i.e. electrons, photons, neutrons, protons, etc. We have demonstrated that the dose results compare favorably with EGS for a photon beam, and with experiment for a high energy neutron beam. Orange is much faster than even the mesh tally options in mcnpX.

Orange is applicable for almost any situation where one wants to calculate the dose.

5 REFERENCES

1. Venselaar J L M, Welleweerd J and Mijnheer B J 2001 Tolerances for photon beam dose calculations using a treatment planning system *Radiother. Oncol.* **60** 191 – 202;
   N. Reynaert et al., "Monte Carlo treatment planning for photon and electron beams," to be subm. to Phys. Med. Biol.

2. Zamenhof R, Redmond E 2nd, Solares G, Katz D, Riley K, Kiger S, Harling O., "Monte Carlo-based treatment planning for boron neutron capture therapy using custom designed models automatically generated from CT data," Int J Radiat Oncol Biol Phys **35** (1996) 383–397

3. Goorly T, McKinney G, Adams K and Estes G 1998 MCNP speed advances in BNCT *Proc. American Nuclear Society Radiation Protection and Shielding Conf, volume II* 1998

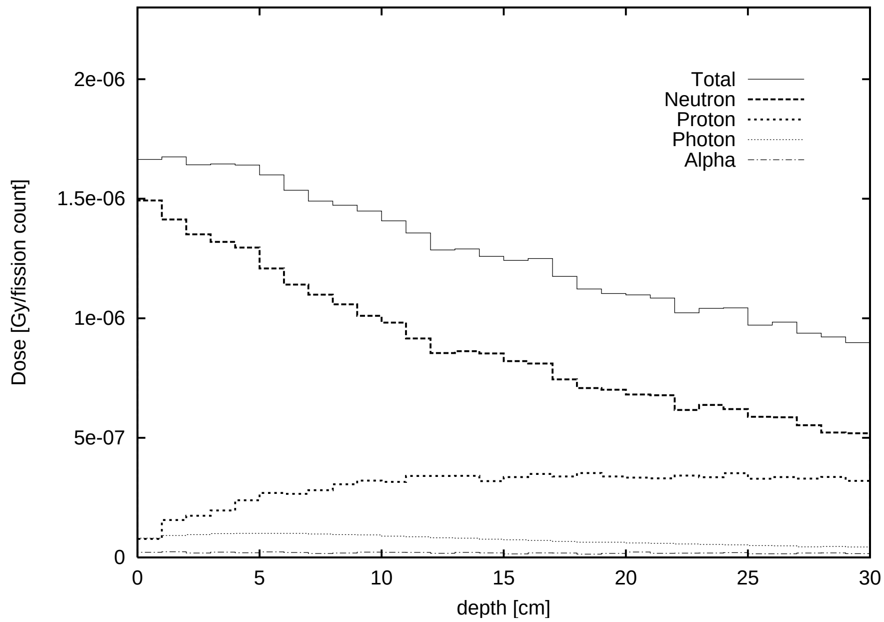

Figure 3: The contribution to the dose from various particle types, for the high energy neutron beam experiment, as calculated by Orange.

Goorly T, McKinney G, Adams K and Estes G 2001 MCNP enhancements, parallel computing and error analysis for BNCT *Frontiers in Neutron Capture Therapy*,599 – 604 (New York: Plenum Publishers)

4. J.T. Goorley, W.S. Kriger, and R.G. Zamenhof “Reference dosimetry calculations for neutron capture therapy with comparison of analytical and voxel models,” Med. Phys. 29 (2002) 145–156

5. Van der Marck S C and Hogenbirk A 2002 ORANGE, a Monte Carlo dose engine for BNCT *Proc. of the 10th Int. Congress of Neutron Capture Therapy* (Bologna: Monduzzi)

6. S.C. van der Marck and A. Hogenbirk, “ORANGE, a new, fast dose engine for radiotherapy treatment planning”, *Proceedings of ICRS-10/RPS-2004*, Madeira, Portugal, 9-14 May 2004

7. W. van der Zee, A. Hogenbirk, and S.C. van der Marck, “ORANGE, a Monte Carlo dose engine for radiotherapy”, subm. to Phys. Med. Biol.

8. Briesmeister J F 1993 MCNP - a general Monte Carlo *n*-particle transport code, Version 4A *Technical Report* LA-12625-M (Los Alamos NM: Los Alamos National Laboratory) Hendricks J S 2001 MCNP4C3 *Technical Report* X-5:RN(U)-JSH-01-17 (Los Alamos NM: Los Alamos National Laboratory)

9. Hendricks J.S. et al., “MCNPX Extensions, version 2.5.e”, Los Alamos report LA-UR-04-0570 (2004)

10. Nelson W R, Hirayama H and Rogers D W O 1985 The EGS4 code system. *Technical Report* SLAC Report 265 (Stanford: Stanford Linear Accelerator Center)

11. Salvat F., Fernández-Varea J.M., and Sempau J., *Penelope – a code system for Monte Carlo simulation of electron and photon transport*, Proc. Penelope-2003 Workshop, Issy-les-Moulineaux, France (NEA, 2003)
12. M.R. Sutton “High-energy neutron dosimetry” *PhD thesis*, Georgia Institute of Technology, 2001
13. Rogers D W O and Mohan R 2000 Questions for comparison of clinical Monte Carlo codes *The use of computers in radiation therapy* (Heidelberg: Springer Verlag)