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Issues with the use of neutron transport codes

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Introduction

Neutron transport codes are very powerful but very complex.

I'll give a couple of examples where their behaviour has been found to differ from what you might expect.

Disclaimer:

- I'm not saying that these are the most important issues.
- I'm not warning about or recommending any particular code.

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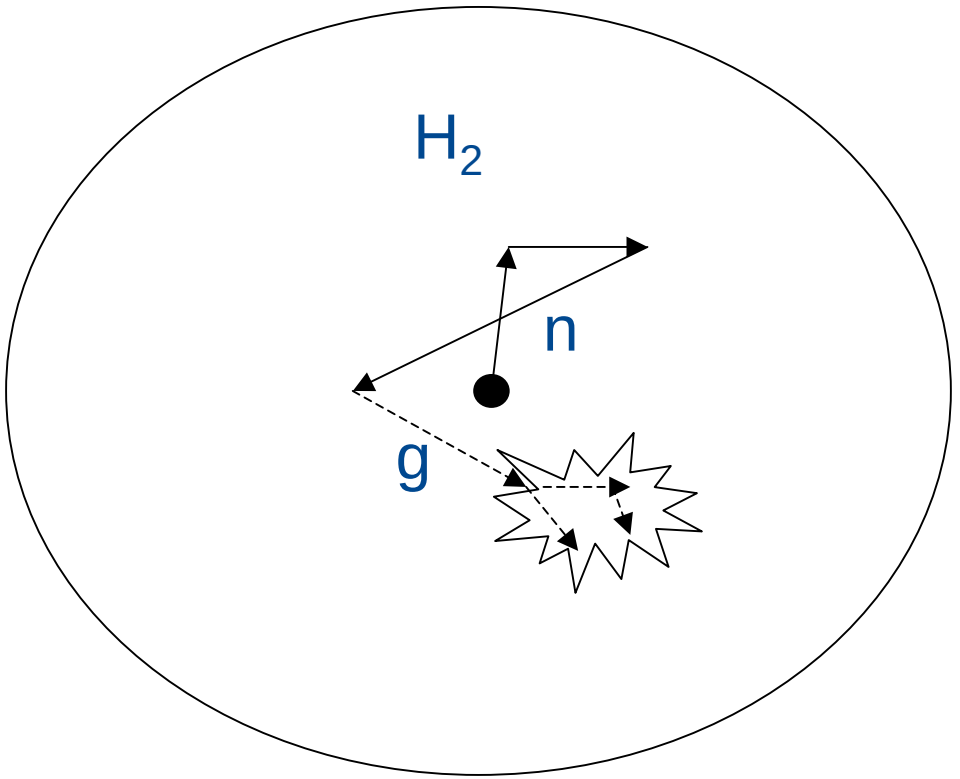
1. Pulse Height tally in MCNPX

- Pulse Height tally with neutrons (F8:n) is allowed in MCNPX.
- We know you must keep everything analogue:
  - Turn off variance reduction
  - Force fully analogue neutron capture
- Yet things can still go wrong

MCNPX: F8 Pulse Height tally



Simple problem:

- Neutrons at a single energy E_i produced at the centre of a huge volume of hydrogen.
- All neutrons are eventually captured to give a 2.2 MeV gamma.
- All the gammas are also absorbed in the hydrogen.

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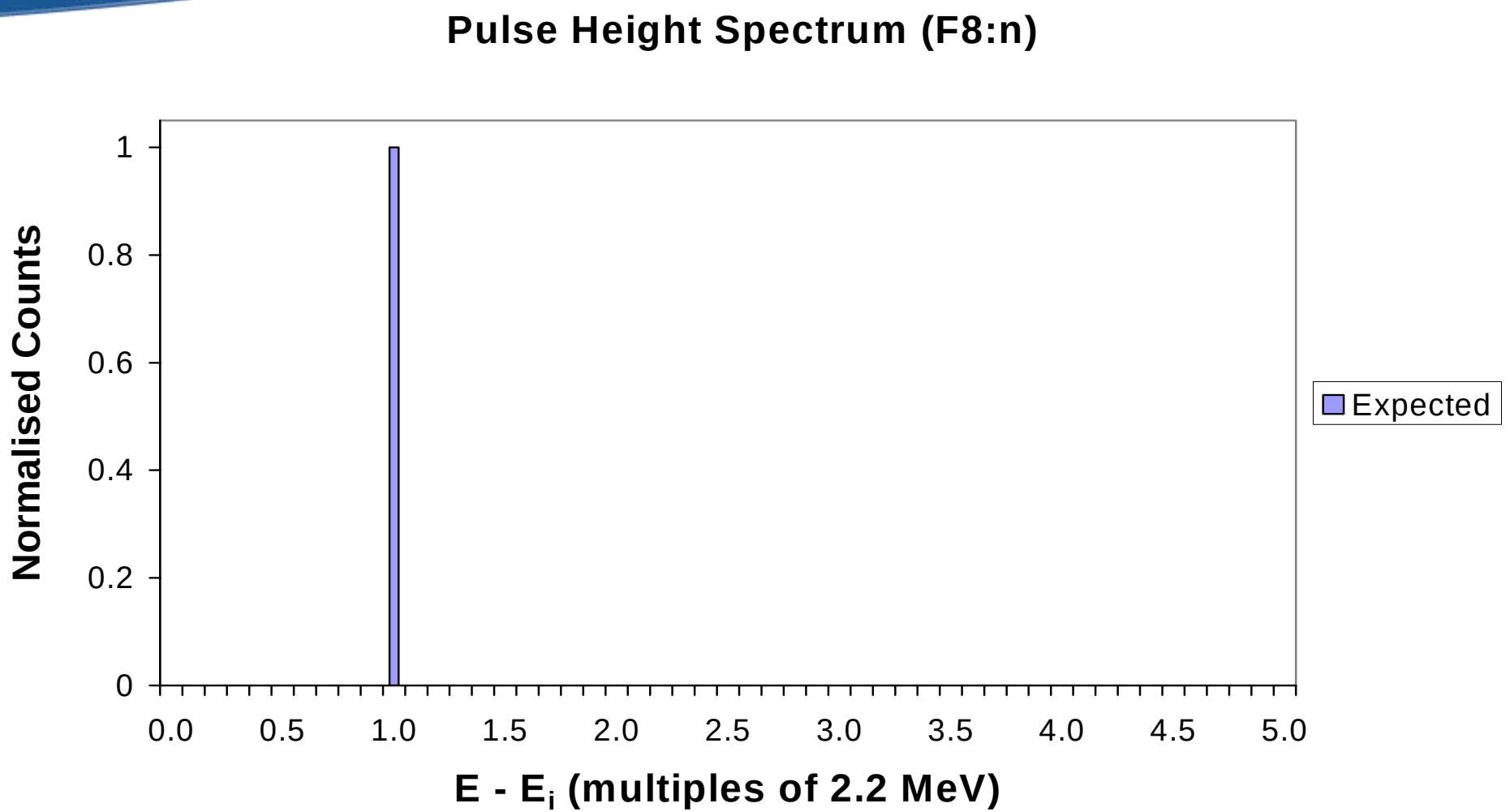
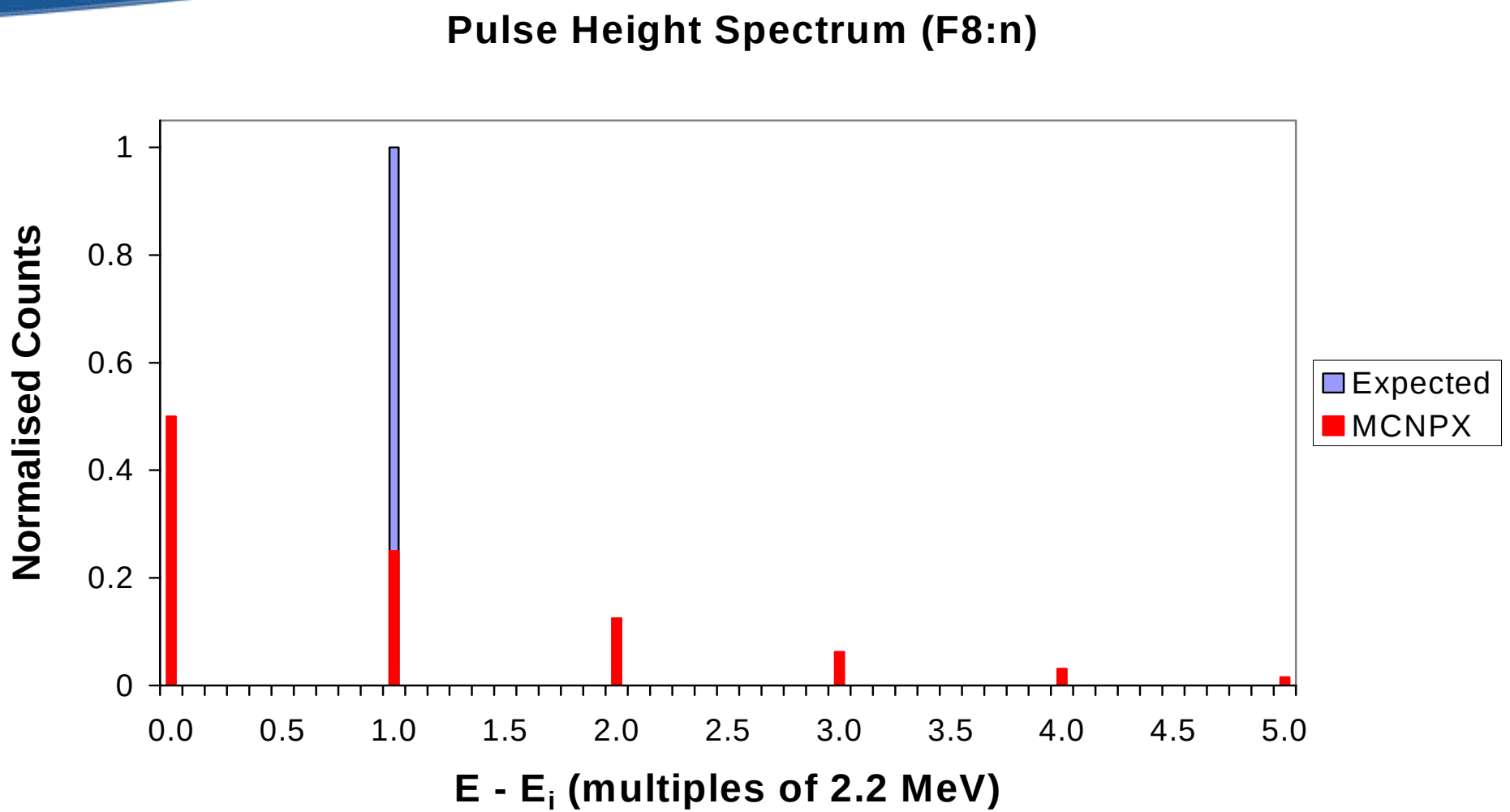
MCNPX: F8 Pulse Height tally



MCNPX: F8 Pulse Height tally



MCNPX: F8 Pulse Height tally

What went wrong?

- Gamma production not correlated with fate of neutron.
- A fundamental property of the libraries – can't easily be changed.
- Also happens with some other reactions ((n, n'), (n, 2n), (p, n), …).
- Some reactions OK ((gamma, n), (gamma, p), …).

Moral: be careful with F8.

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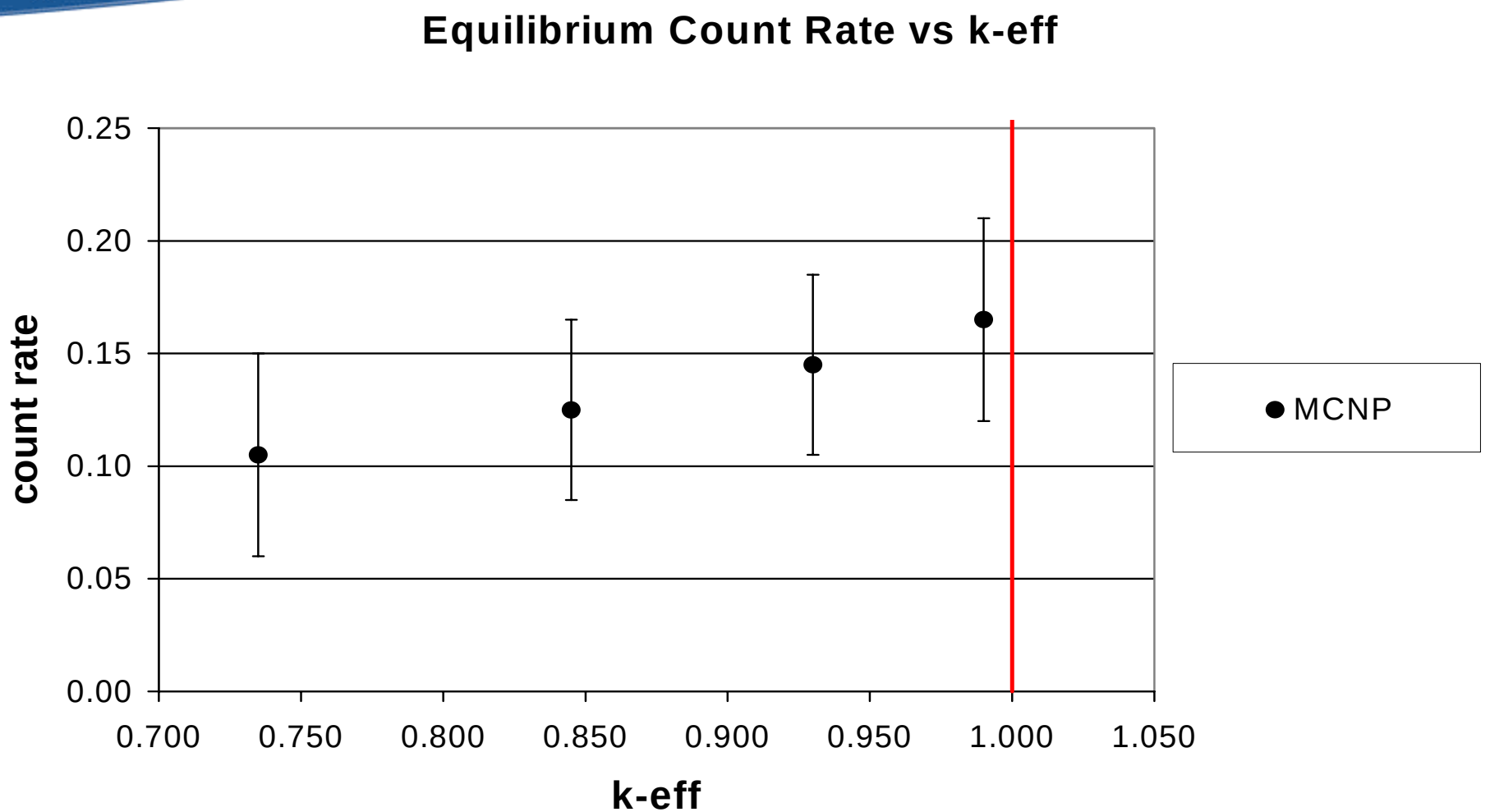
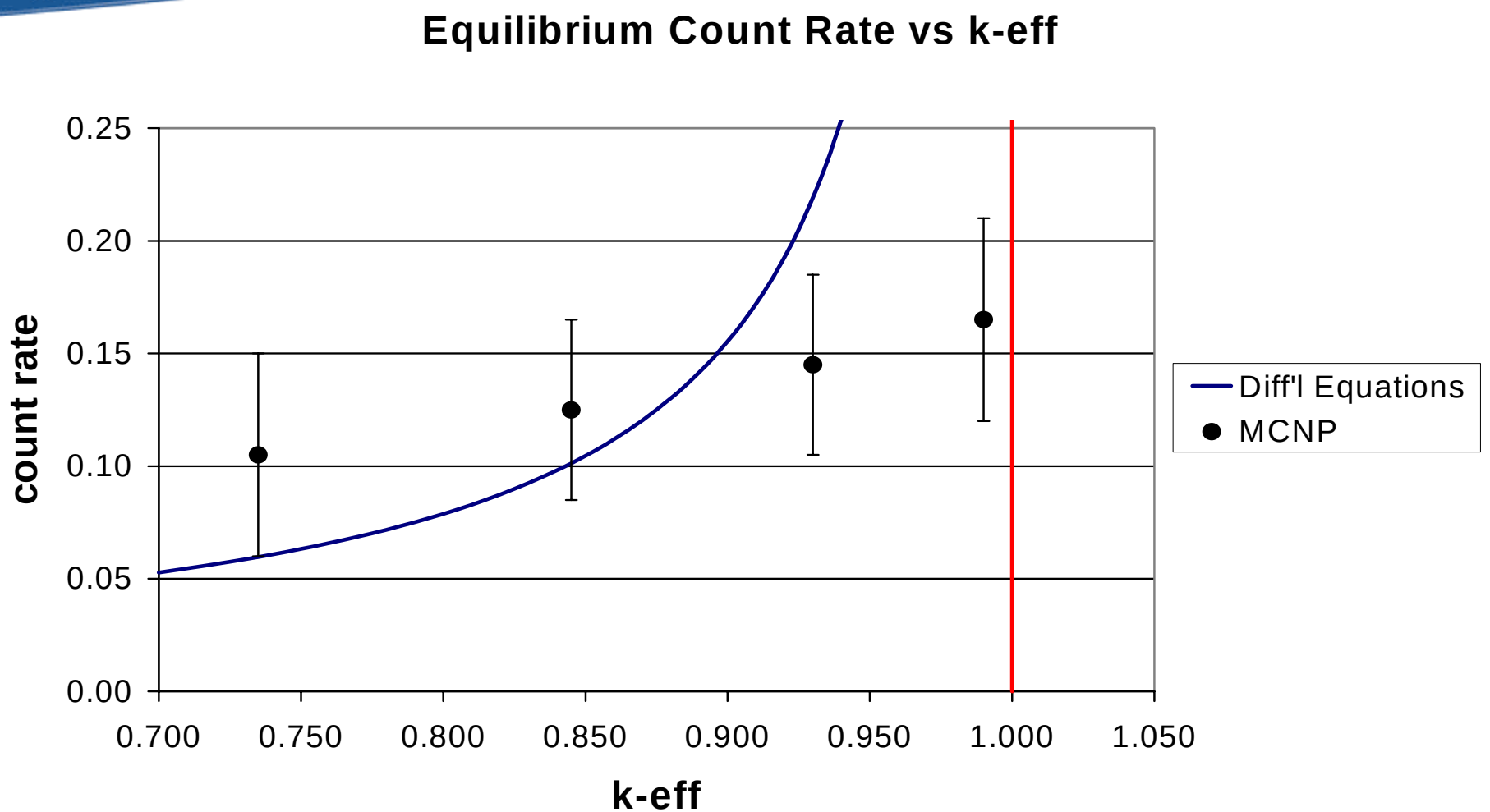
2. MCNP near critical

- Neutron detector near a sub-critical assembly + external source.
- Want to know how the count rate varies with k-eff (ratio of neutrons produced in one generation to those used up in the previous generation).
- Repeatedly increase the k-eff a bit (by changing a control rod) and calculate the final equilibrium count rate each time.
- On the real system, measure the count rate and read off the k-eff.

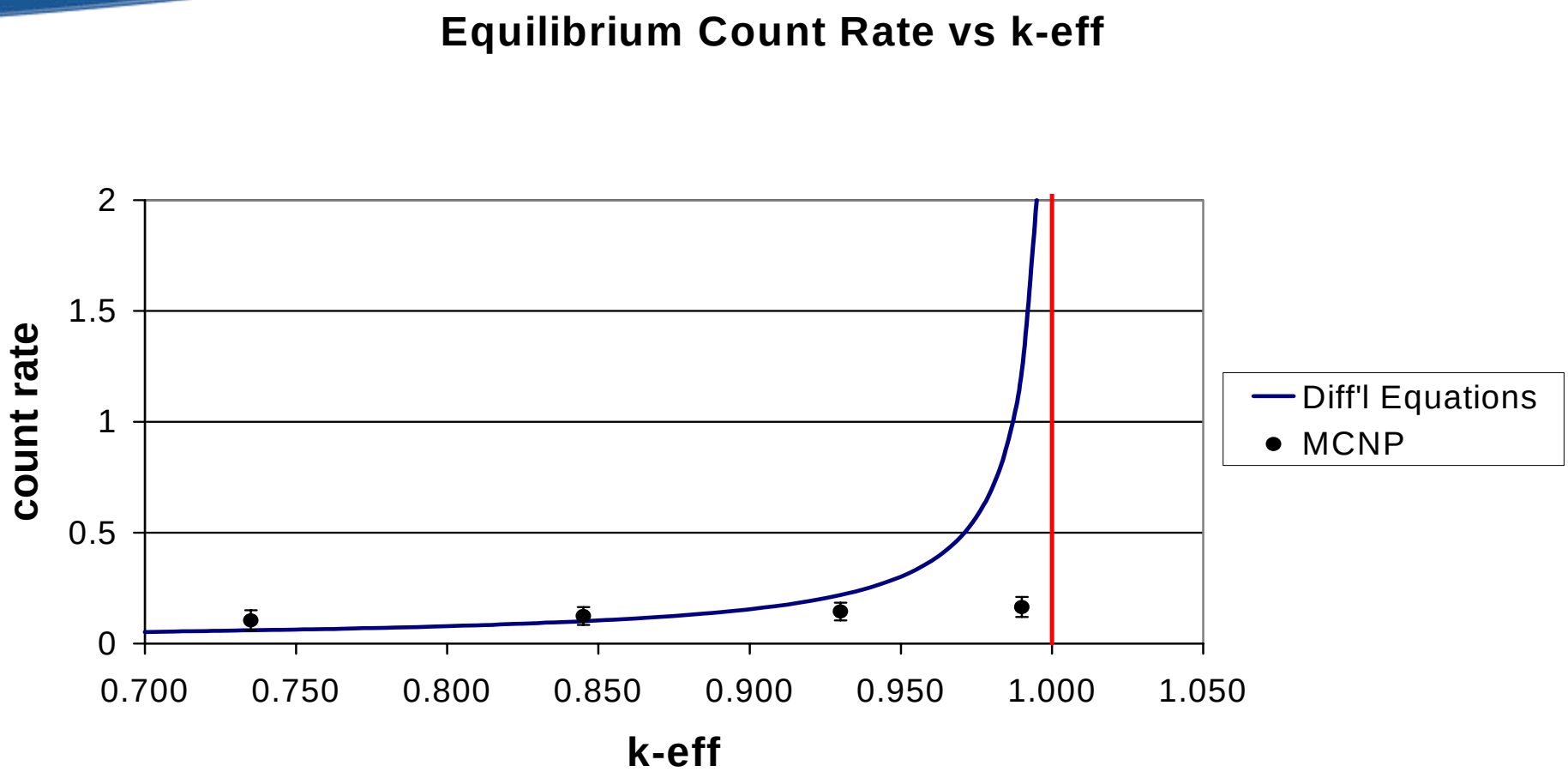
MCNP: Near critical



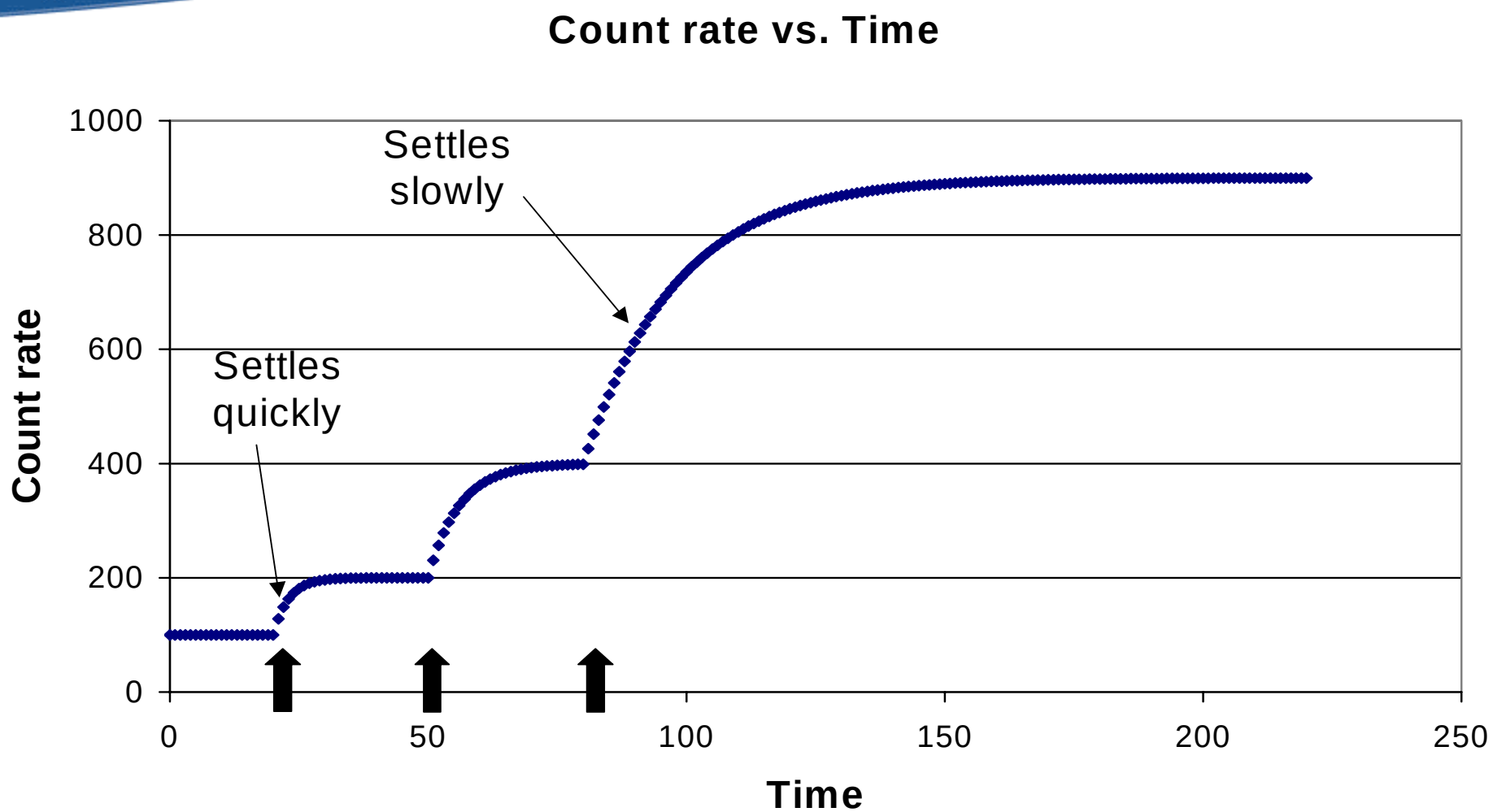
MCNP: Near critical



MCNP: Near critical



MCNP: Near critical



MCNP: Near critical

What went wrong?

- Near critical, the system can take several minutes to settle.
- This is 10^{10} MCNP time steps.
- Histories were cut off before the true population was built up.

Moral:

- Start some of the neutrons from the fuel, not just from the external source.
- Watch for warnings.

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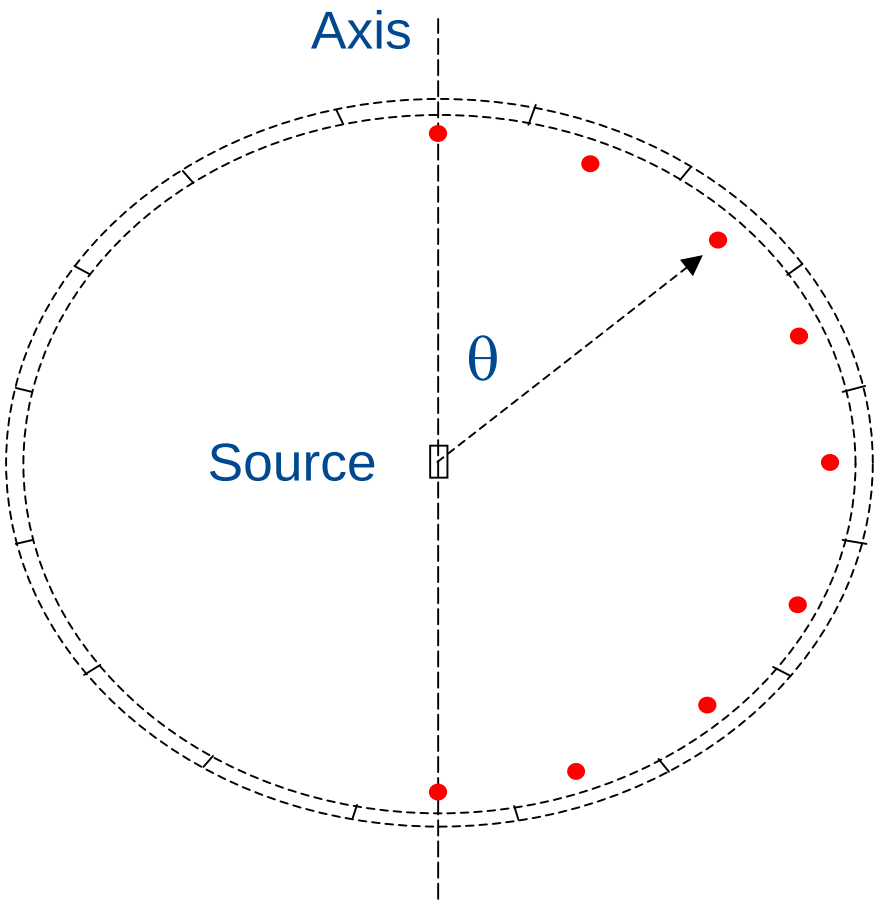
3. Point Estimators in MCBEND

- Neutron sources have internal structure and do not emit equally in all directions.

- We sometimes want to calculate this anisotropy for a particular source type.

- We have used MCNP in the past, but recently started using MCBEND as well.

MCBEND: Point estimators



Geometry:

•Source is cylindrical with an axis of symmetry.

•For scoring, set up a thin spherical shell in the vacuum around the source.

•Divide the shell into regions and score the fluence in each region.

•Also set up ‘point estimators’, which give the fluence at a point directly, without averaging over a volume.

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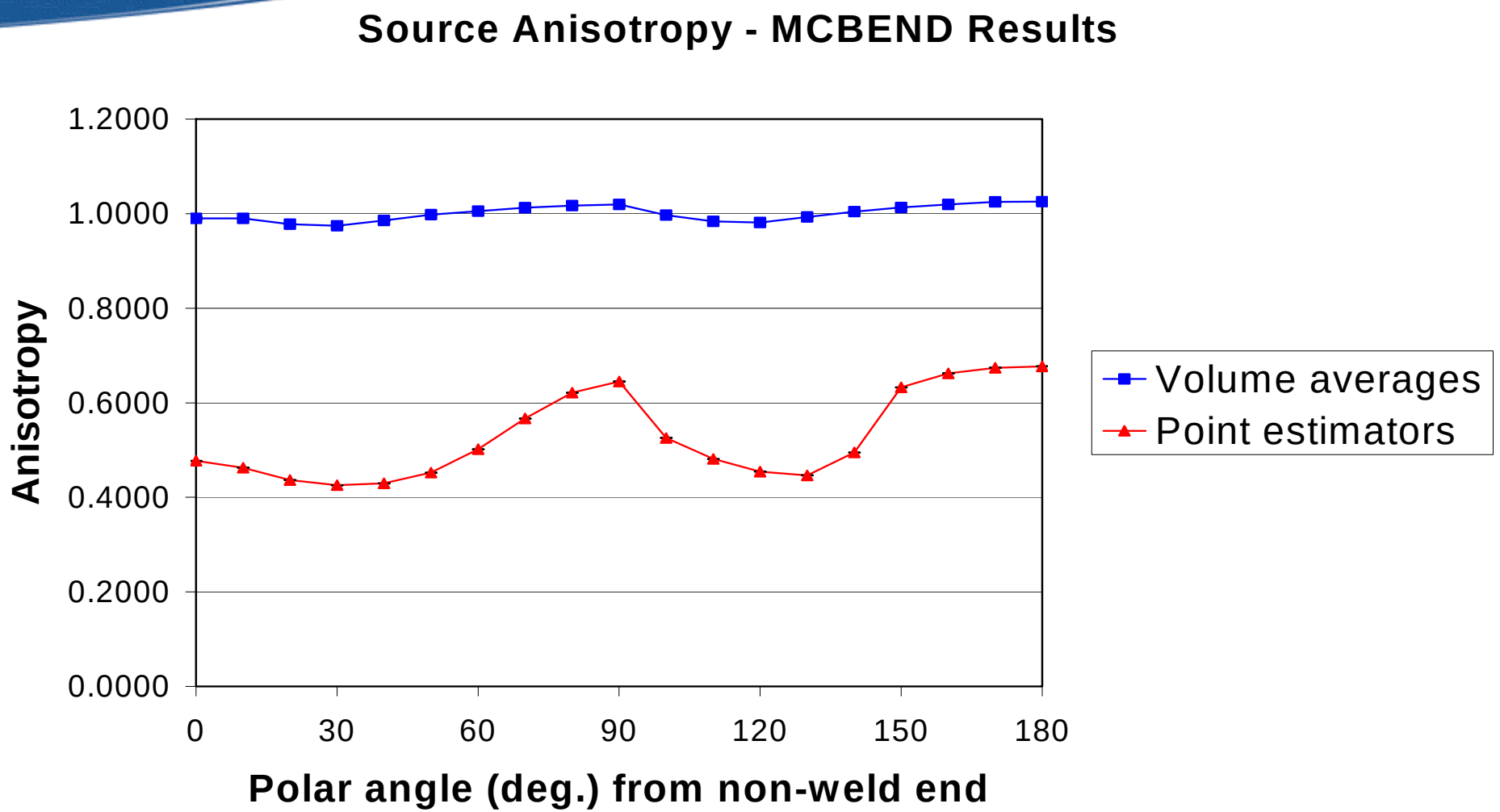
MCBEND: Point estimators



What went wrong?

- Anisotropies are only a few percent, so we need a large number of histories (50 million).
- The Point Estimator module is single precision.
- The later histories stopped making any difference to the accumulating total.

Moral:

- Do a sanity check on the results.
- Calculate more than one way if possible.
- Hope that the double precision version appears soon!

Conclusions

"Trust no-one, Mr. Mulder"

– Well-Manicured Man, The X-Files