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Title: Multigroup Cross-section Generation in MCNP6.3

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Multigroup Cross-section Generation with MCNP6.3

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Motivation

Motivation

- Historically, MCNP development has not been focused on nuclear reactor applications.
- Recent institutional investments through the LANL Laboratory Directed Research & Development (LDRD) Program have focused on developing capabilities for nuclear reactor applications.
- **A new special tally treatment for multigroup cross section calculations is in production for MCNP6.3**
- Work is continuing in this area, with a correct focus on alternative tracking algorithms (i.e., Delta tracking), improved energy deposition and burn-up/activation tallies, and new tooling to support efficient workflows for nuclear reactor applications

Background (1)

Multigroup cross sections are typically needed in deterministic transport and diffusion codes for reactor physics applications

- Accurate multigroup cross sections require the use of an appropriate weighting spectrum
- The weighting spectrum should be representative of the application that the multigroup cross sections are being used for

In simplified notation, the multigroup cross section can be computed as a ratio integrals,

$$\bar{\Sigma}_{x,g} = \frac{\langle \Sigma_x, \phi \rangle_g}{\langle \phi \rangle_g}, \text{ where}$$

- $\bar{\Sigma}$ multigroup cross section
- Σ continuous-energy cross section
- ϕ weighting spectrum
- x reaction channel
- g incident-energy group
- $\langle a, b \rangle$ the inner product of a and b integrated over all phase space

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Background (2)

While the multigroup reaction cross sections are straightforward, the scattering angle/energy and fission energy terms are slightly more involved and limited (analog transport is used)

The multigroup Legendre moment, l, for the scattering matrix defining transitions from group g' to g is

$$\bar{\Sigma}_{sl,g' \to g} = \frac{\langle \Sigma_{sl}, \phi \rangle_{g' \to g}}{\langle \phi \rangle_{g'}}.$$

The multigroup fission neutron spectra is

$$\bar{\chi}_g = \frac{\sum_{g'=1}^{G} \left\langle \nu \Sigma_f, \phi \right\rangle_{g' \to g}}{\sum_{g=1}^{G} \sum_{g'=1}^{G} \left\langle \nu \Sigma_f, \phi \right\rangle_{g' \to g}},$$

which produces a normalized fission neutron energy spectrum. To separately obtain prompt and delayed fission neutron energy spectra, the integrals are binned by time.

Multigroup Cross Section Tally Options

Four new tally special treatment options (`FT` card) have been added to assist with reactor analyses:

- `SPM` Collision exit energy-angle scatter probability matrices
- `MGC` Flux weighted multigroup cross sections
- `FNS` Induced fission neutron spectra
- `LCS` Legendre coefficients for scatter reactions

These new multigroup tally capabilities have been thoroughly described and verified via code-to-code comparisons [1].

Flux-weighted Multigroup Cross Sections

`FTn MGC fg`

- `MGC` Flux weighted multigroup cross sections
- `fg` Flag for microscopic (barns) or macroscopic (1/cm) cross section calculation

Description of the Multiplier Bins for the `MGC FT` Option.

Bin #	Units	Values
1	$n/(cm^2 \cdot s)$	Flux (used as a divisor for the other bins)
2	sh/cm	Inverse velocity
3	barns	Total cross section
4	barns	Absorption cross section
5	barns	Fission cross section
6	barns	Total or prompt fission production cross section
7	barns	Delayed fission production cross section
8	barns	Fission heat production cross section
9	barns	Capture cross section (Absorption + Fission)
10	barns	Scatter cross section [Total − (Absorption + Fission)]

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Flux-weighted Scattering Matrices and Fission Spectra

Multigroup scattering matrix options

`FTn SPM na` (cosine-binned scattering matrices)

- `SPM` Collision exit energy-angle scatter probability matrices
- `na` Integer number of equally-spaced cosine bins

`FTn LCS lo` (Legendre coefficient scattering matrices)

- `LCS` Legendre coefficients for scatter reactions
- `lo` Integer number of maximum Legendre scattering order

Multigroup fission energy spectra

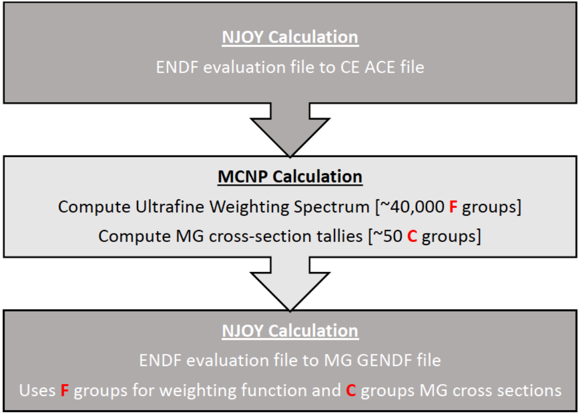
`FTn FNS nt`

- `FNS` Induced fission neutron spectra
- `nt` Integer number of delayed neutron time bins
  - If `nt` is not specified, then a `T` card needs to be used to specify time binning to separate various prompt and delayed neutrons emitted from fission.

Code-to-code Verification Efforts (1)

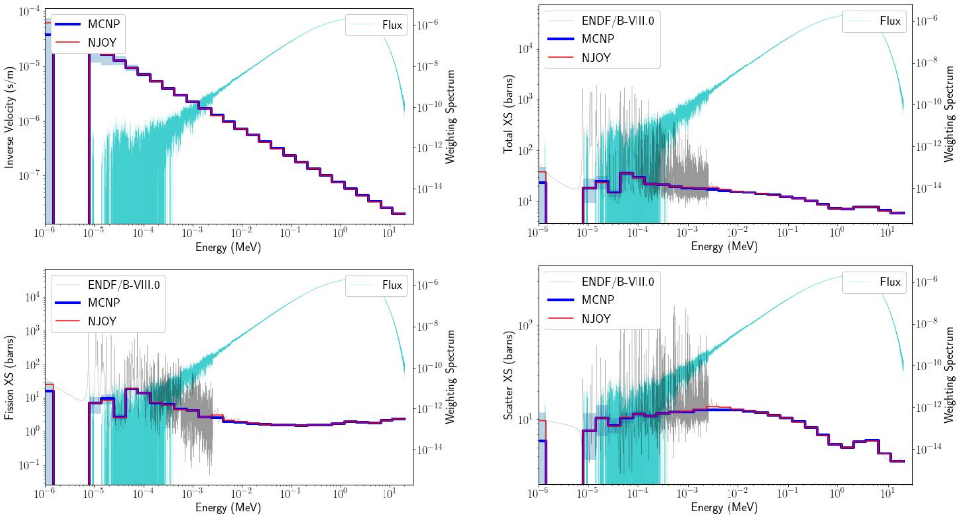
- Compared new multigroup special treatments to those produced using NJOY
  - Calculated a fine-group energy weighting spectrum with MCNP
  - Inserted the weighting spectrum into NJOY

MCNP and NJOY Multigroup Comparison



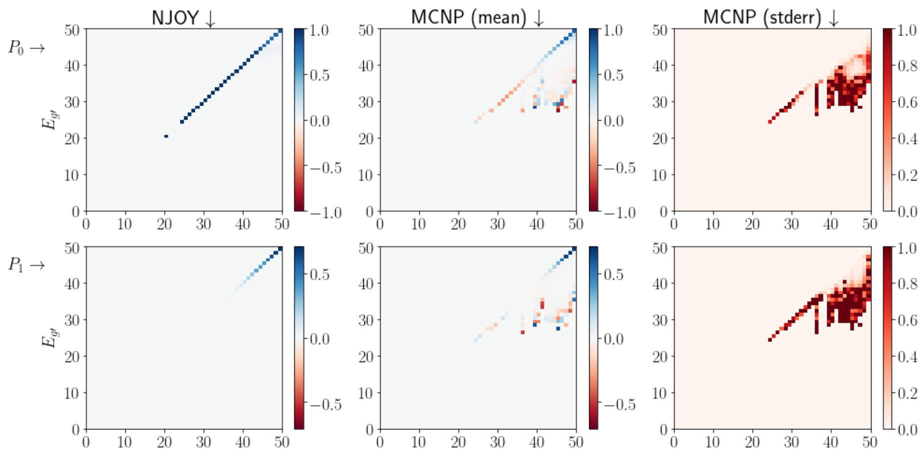
Code-to-code Verification Efforts (2)

Multigroup Cross Section `MGC` Option Verification



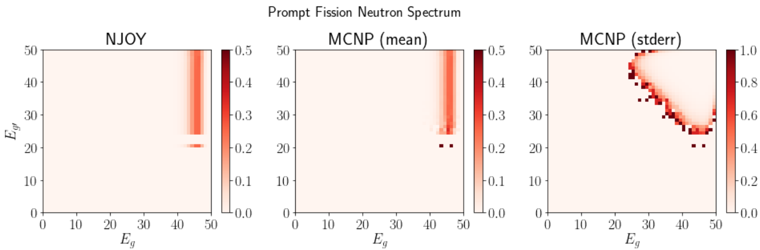
Code-to-code Verification Efforts (3)

Legendre Scattering Coefficient `LCS` Option Verification

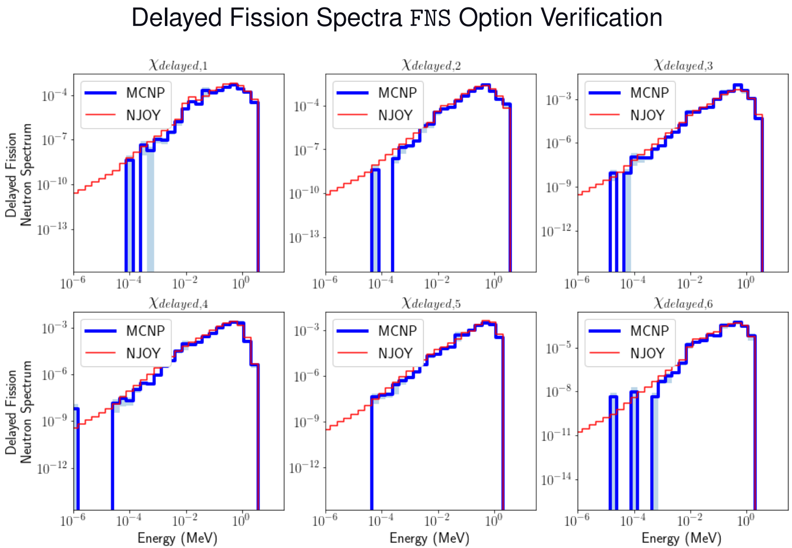


Code-to-code Verification Efforts (4)

Prompt Fission Spectra `FNS` Option Verification



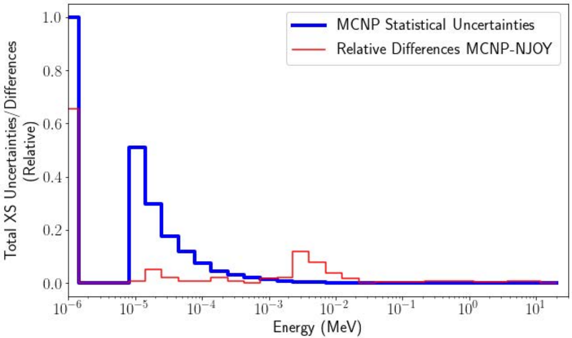
Code-to-code Verification Efforts (5)



Code-to-code Verification Efforts (6)

- The Monte Carlo statistical uncertainties, especially for the scattering angle/energy matrices and the fission energy spectra tallies that are required to use analog transport, are challenging to overcome
- For the multigroup reaction cross sections, differences are typically smaller or equal to the statistical uncertainties... except in the vicinity of the unresolved resonance region

Differences in MCNP and NJOY `MGC` Total Cross Section



Summary

- In comparison to the NJOY-produced multigroup cross sections, the MCNP-produced multigroup cross sections are generally consistent
  - Statistical uncertainties are challenging
  - The unresolved resonance region may be looked at in the future
- The `SPM` and `LCS` options were compared to each other for internal consistency
- Some reactor pin-cell-like problems were used to compare to multigroup capabilities in other Monte Carlo codes (e.g. Serpent, OpenMC)

Questions?

Backup Slides

References

[1] R. B. Wilkerson, G. McKinney, M. E. Rising, and J. A. Kulesza, "MCNP Reactor Multigroup Tally Options Verification," Tech. Rep. LA-UR-20-27819, Los Alamos National Laboratory, Oct. 2020.