Development of Multigroup Cross Section Generation Code MC^2-3 for Fast Reactor Analysis

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Background

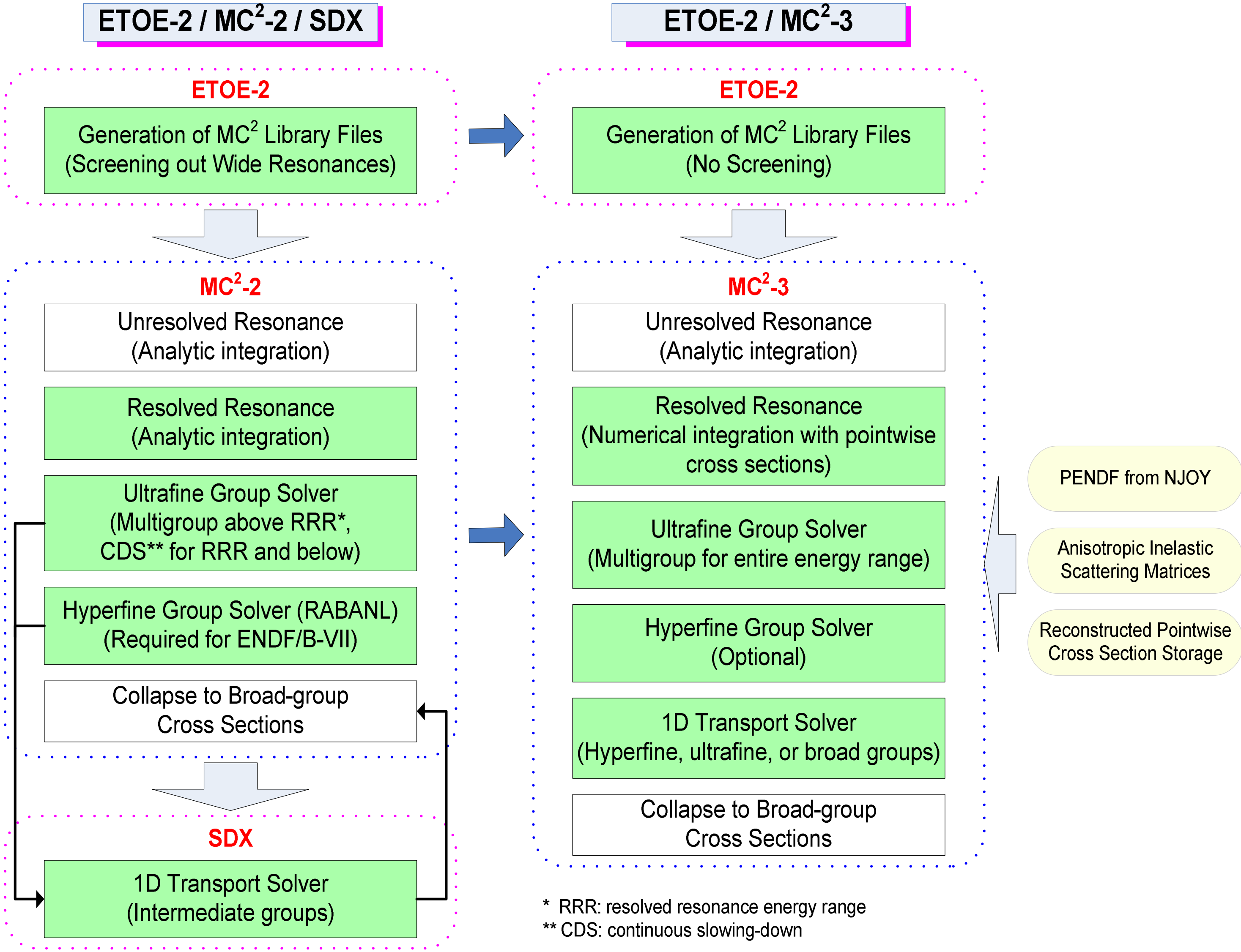
- Under the Nuclear Energy Advanced Modeling and Simulation (NEAMS) of U.S. DOE, an integrated, advanced neutronics code system is being developed to allow the high fidelity description of a nuclear reactor and simplify the multi-step design process
  - Development of UNIC with unstructured finite element mesh capabilities on a large scale of parallel computation environment
  - Integration with thermal-hydraulics and structural mechanics calculations

- As part of this effort, an advanced multigroup cross section generation code named MC^2-3 is being developed
  - The ANL multigroup generation code system, ETOE-2 / MC^2-2 / SDX, has been successfully used for fast reactor analysis
  - Recent studies with the ENDF/B-VII.0 data identified some improvement needs of MC^2-2
    - Increased importance of resolved resonances in the ENDF/B-VII.0 data due to the extended upper energy cutoff and significantly increased number of resolved resonances required the use of RABANL for a rigorous treatment of resolved resonances
    - Use of RABANL is limited to the relatively low energy range where the isotropic source approximation is valid

ETOE-2 / MC^2-2 / SDX

- ETOE-2
  - Generate MC^2 libraries by processing ENDF/B data, including ultrafine group smooth cross sections (2,082 groups with constant lethargy from 20 MeV to 0.4 eV)
  - Screen out wide resonances to smooth cross sections
  - Convert the resolved resonances in the Reich-Moore formalism to those in the multi-pole formalism
- MC^2-2
  - Self-shield unresolved and resolved resonances using the generalized resonance integral method based on the narrow resonance (NR) approximation
  - Perform the consistent P1 or B1 transport spectrum calculations
    - Multigroup method for above resolved resonance energy range
    - Continuous slowing down method for the resolved resonance energy range
  - RABANL option for the hyperfine group slowing-down calculation based on isotropic elastic scattering (applicable below ~tens keV)
- SDX
  - Perform the 1D integral transport calculation to account for the local heterogeneity effects

MC²-2/SDX vs. MC²-3



* RRR: resolved resonance energy range
** CDS: continuous slowing-down

Changes and Improvements in MC^2-3

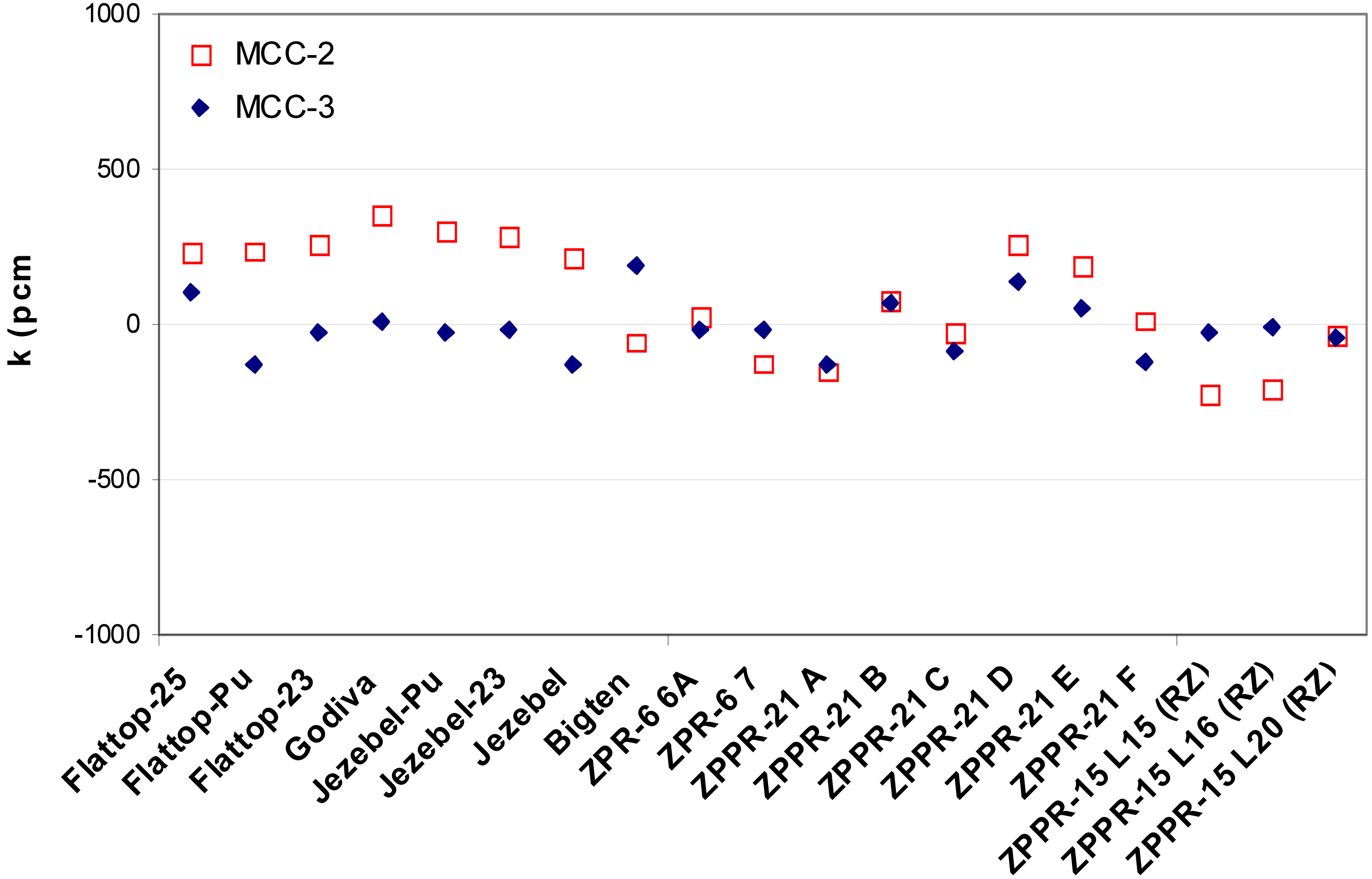
- Numerical integration of resolved resonances with pointwise cross sections based on the NR approximation
  - Reconstruction of pointwise cross sections with Doppler broadening
  - Optionally, use of PENDF files from NJOY
- Multigroup spectrum calculation with the consistent P_1 transport equation for the entire energy range
- New capability of treating anisotropic inelastic scattering
- Self-shielding of resonance-like cross sections above the resonance energy for intermediate-weight nuclides (Fe, Cr, Ni, etc.)
- 1D transport calculation with ultrafine (2082) or user-defined groups (SDX capability)
- 1D hyperfine (> ~100,000) group transport calculation
  - MOC solver with higher-order anisotropic scattering in the LS and CMS (up to ~1 MeV)

$$\sigma_{sl}^{i}(g \to g') = \frac{1}{\psi_{lg}} \int_{u_{g'-1}}^{u_{g'}^{*}} du' \int_{u_{g-1}^{*}}^{u_g} du \frac{\psi_l(u)\sigma_s^i(u)e^{-(u'-u)} P_l(\mu_s^i)}{(1-\alpha_i)} \sum_{n=0}^{N} (2n+1) f_n^i(u) P_n(\mu_c^i)$$

- Inline cross section generation as a module of UNIC
  - Standalone version for conventional multi-step analyses
- FORTRAN 90/95 memory structure

Critical Experiments

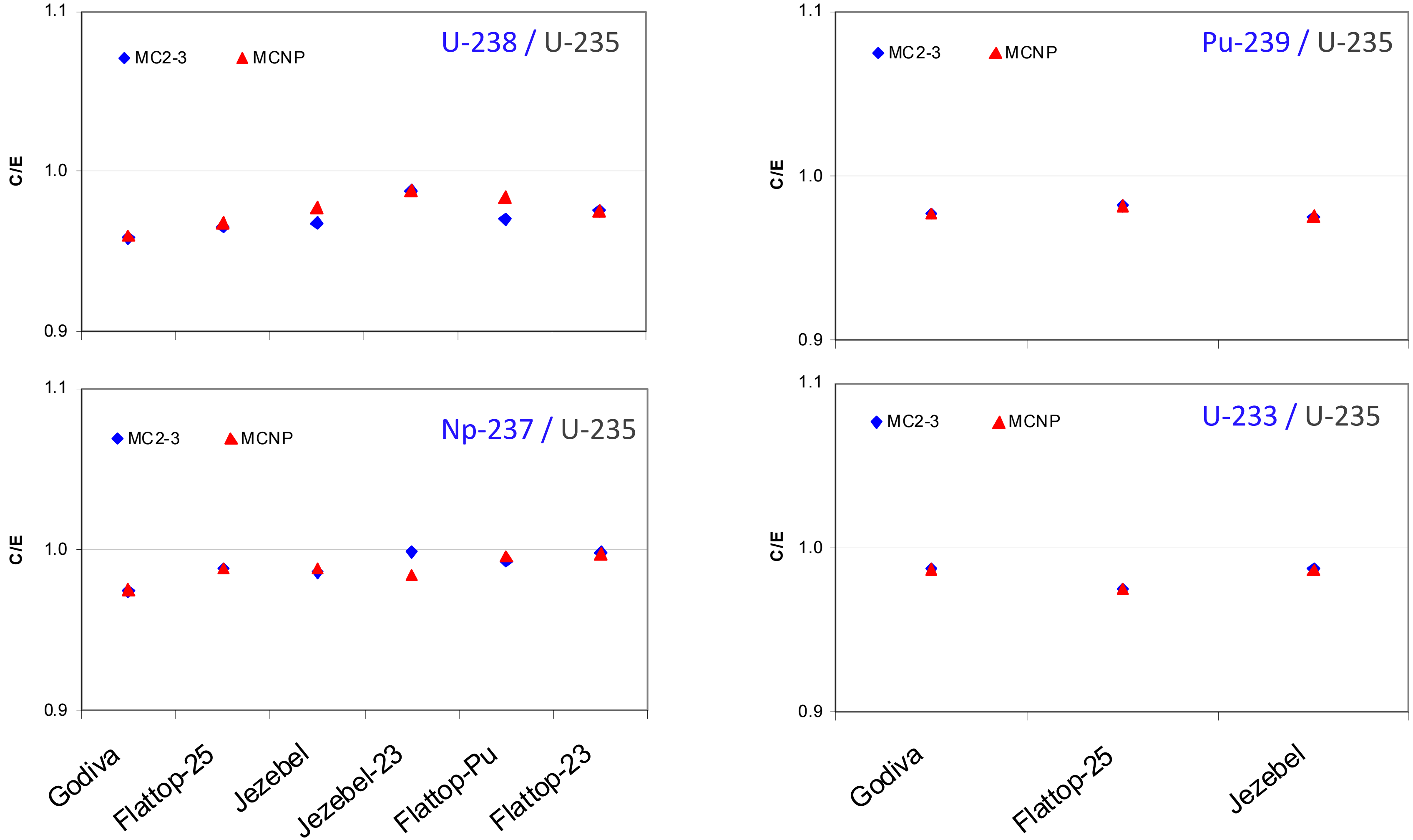
- Δk in pcm from Monte Carlo results



C/E of Fission Reaction Rate Ratios for LANL Assemblies

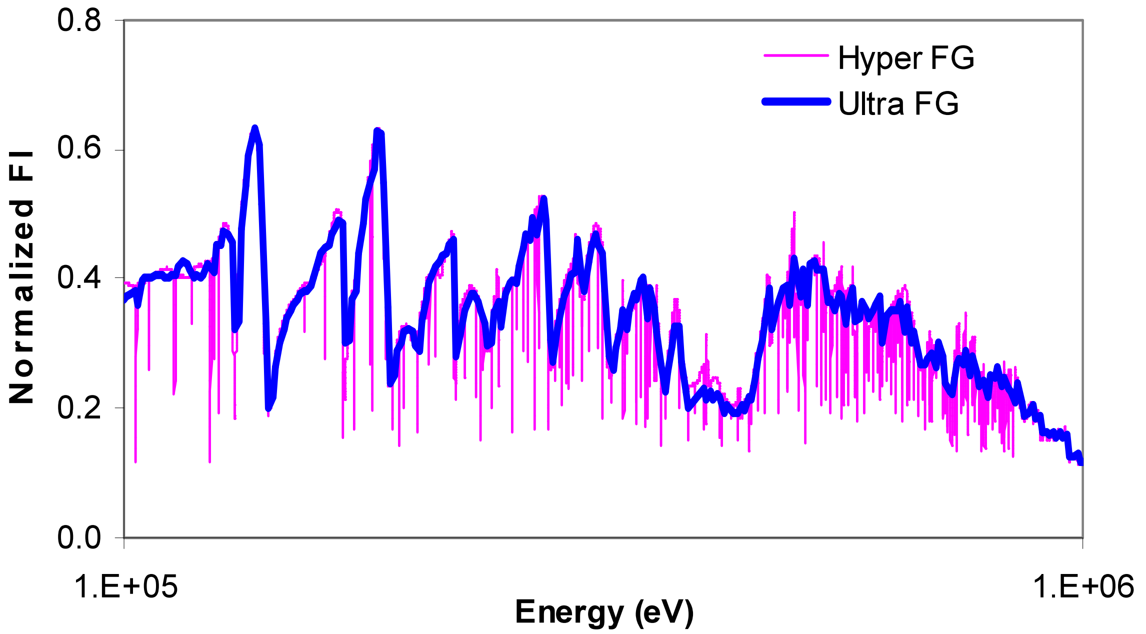
Assembly	Data		$\sigma_f^{U238} / \sigma_f^{U235}$	$\sigma_f^{Np237} / \sigma_f^{U235}$	$\sigma_f^{U233} / \sigma_f^{U235}$	$\sigma_f^{Pu239} / \sigma_f^{U235}$
GODIVA	Experiment		0.1643±0.0018	0.8516±0.013	1.59±0.03	1.4152±0.025
	C/E	MCNP [a)]	0.960	0.975	0.987	0.977
		MC^2-3 [b)]	0.958	0.974	0.987	0.977
JEZEBEL	Experiment		0.2133±0.0023	0.9835±0.016	1.578±0.027	1.4609±0.013
	C/E	MCNP	0.978	0.988	0.986	0.975
		MC^2-3	0.968	0.986	0.987	0.975
JEZEBEL -23	Experiment		0.2131±0.0026	0.9970±0.015		
	C/E	MCNP	0.989	0.984		
		MC^2-3	0.988	0.998		
FLATTOP -25	Experiment		0.1492±0.0016	0.7804±0.01	1.608±0.003	1.3847±0.012
	C/E	MCNP	0.968	0.988	0.975	0.982
		MC^2-3	0.966	0.988	0.975	0.982
FLATTOP -Pu	Experiment		0.1799±0.002	0.8561±0.012		
	C/E	MCNP	0.984	0.996		
		MC^2-3	0.970	0.992		
FLATTOP -23	Experiment		0.1916±0.0021	0.9103±0.013		
	C/E	MCNP	0.976	0.997		
		MC^2-3	0.976	0.998		

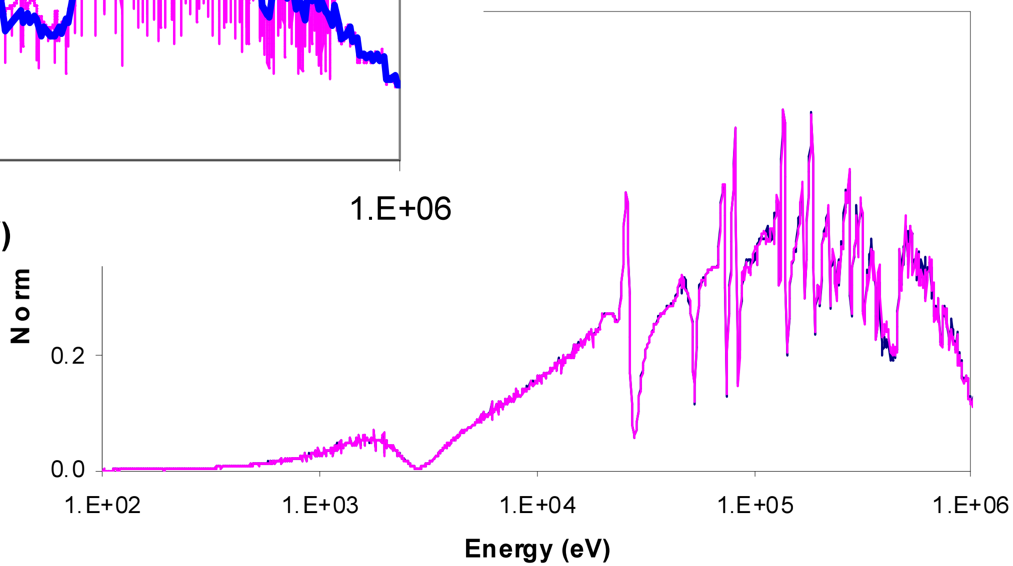
C/E of Fission Reaction Rate Ratios for LANL Assemblies



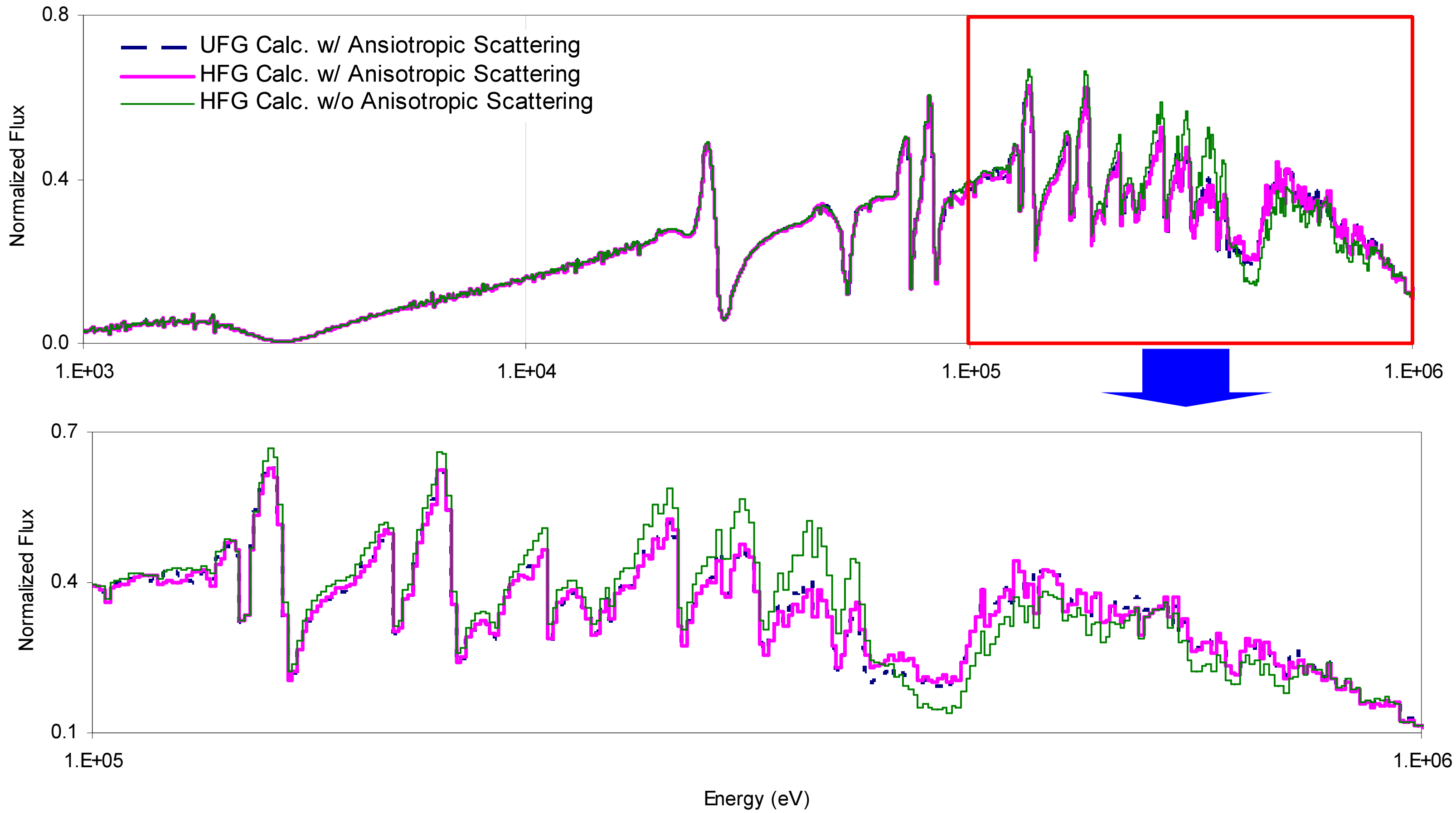
Hyperfine-Group Spectrum Calculation

- Inner core composition of ZPR-6/6A

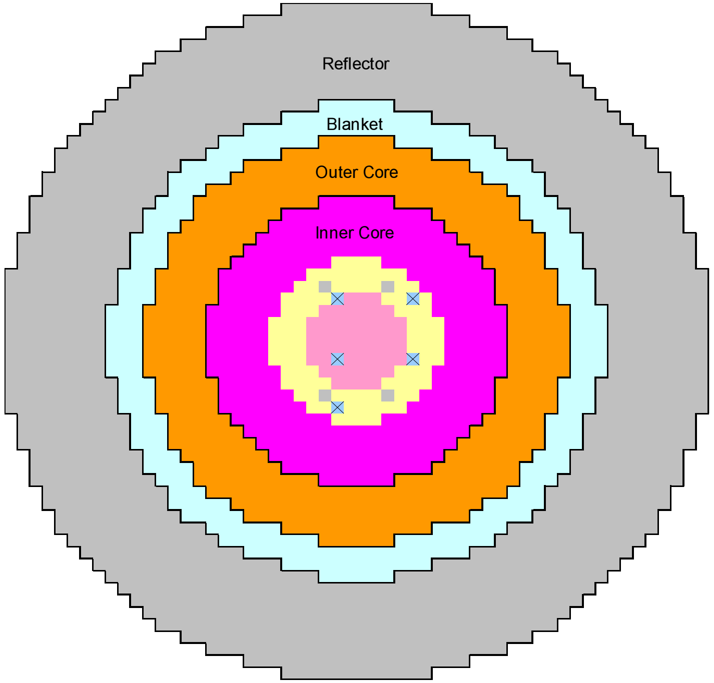


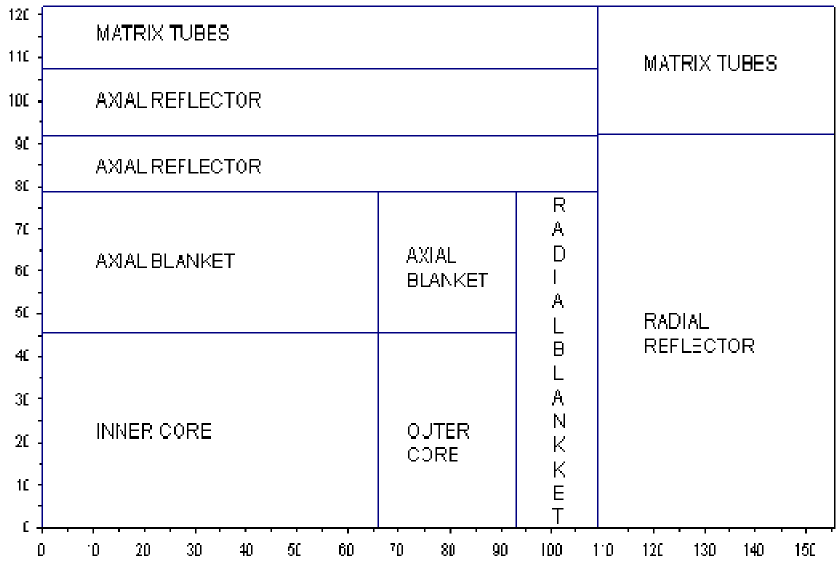


Ultrafine and Hyperfine Group Spectrum Calculation with Anisotropic Scattering Sources



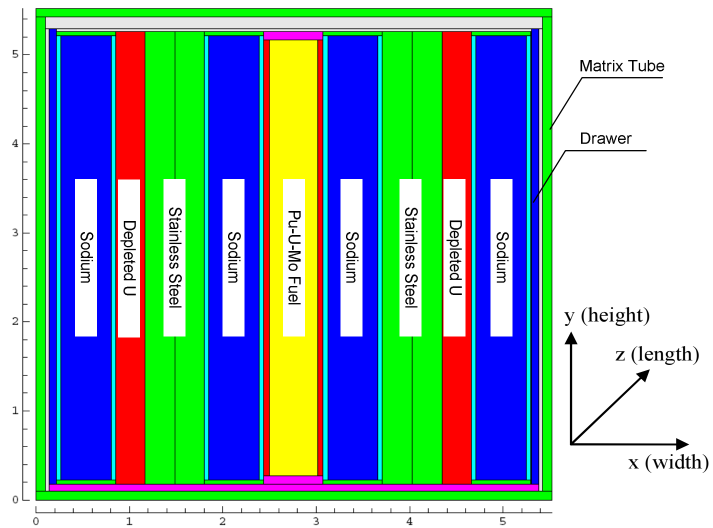
ZPPR-15A Critical Experiments





Loading	Experiment	VIM	MC^2-2	MC^2-3
15	1.00046	0.99985	-392	-245
16	0.99627	0.99571	-393	-244
20	0.99853	0.99742	-316	-192

* Uncertainty: Experiment < ±0.0018, VIM < ±0.00020



ZPR-6 Critical Experiments

A full core heterogeneous reactor calculations with explicit fuel plate representation

- 50,000,000 vertices (~equivalent to 200 million PARTISN finite difference cells)
- 200+ angles with P_5 anisotropic scattering
- 9, 33, 70, and 230 groups
- No thermal-hydraulics considerations (i.e. clean comparison with MCNP/VIM)

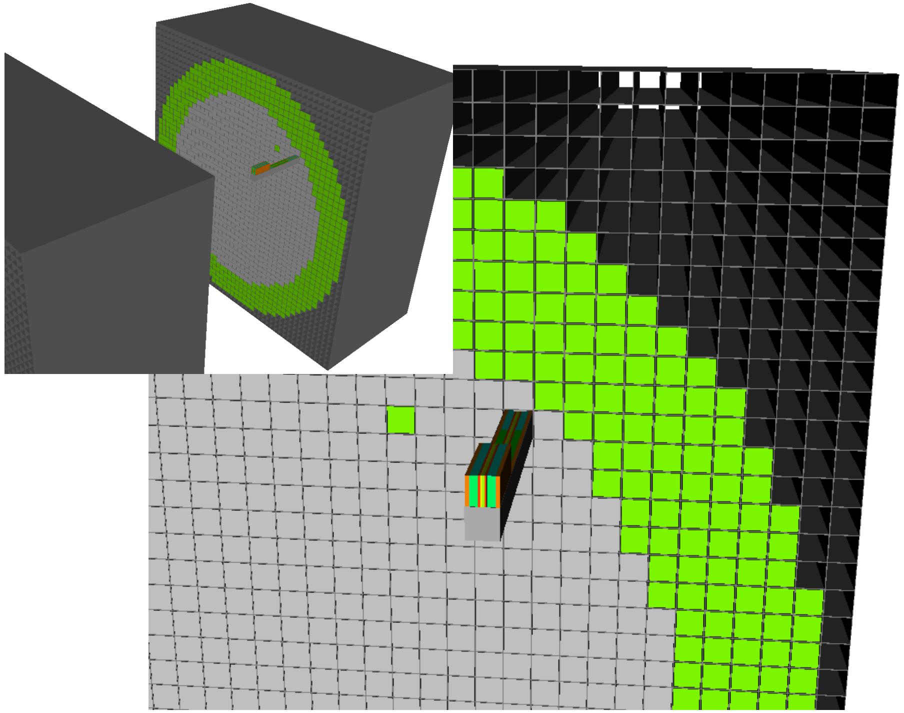
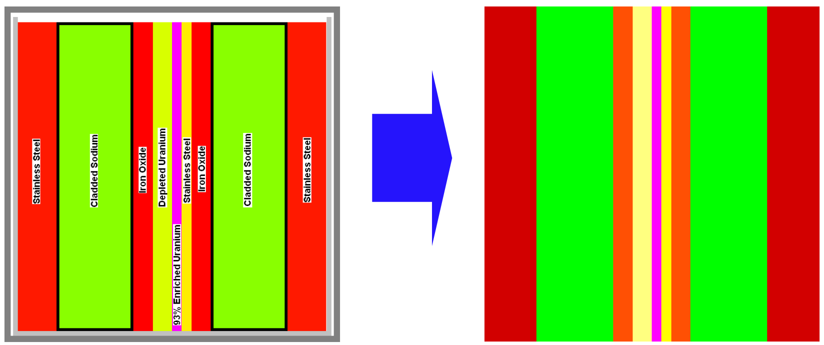


Plate by Plate ZPR Geometry

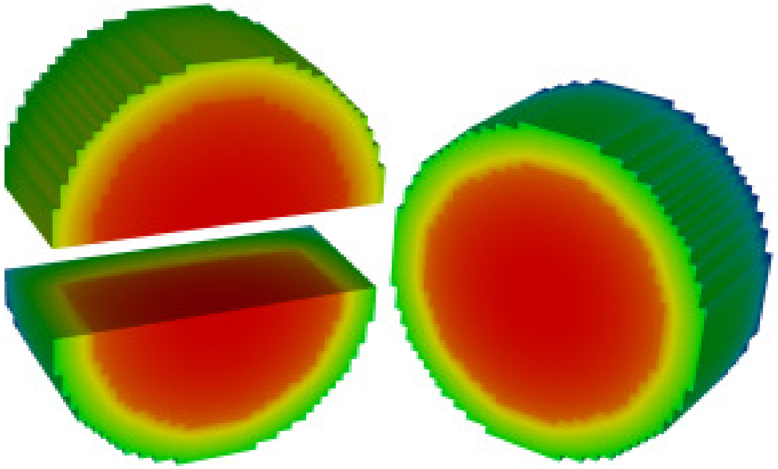


UNIC Results with MC^2-3 Cross Sections

- Homogeneous cell cross sections with MC^2-3 without the heterogeneity effect of fuel drawers

Energy Group	K-effective	Δk pcm
9	0.99513	113
33	0.99373	-27
116	0.99355	-45
230	0.99344	-56

VIM : 0.99400 ±0.00020



Power Distribution

- Cell-averaged cross sections with the 1D slab transport calculation of MC^2-3 to account for the heterogeneity effect of fuel drawers

VIM
0.99981 ±0.00025

Energy Group	K-effective	Δk pcm
9	1.00007	26
33	0.99966	-15
116	0.99965	-16
230	0.99966	-15

Summary

- New multigroup cross section generation code MC^2-3 has been developed with improved methods
- Verification tests with LANL, ZPR-6, ZPPR-15A, ZPPR-21, and BFS critical experiments showed more rigorous and accurate solutions compared to MC^2-2 / SDX
- 1D hyperfine-group transport calculation capability with higher-order anisotropic scattering sources is near completion
- Initial integration of MC^2-3 into UNIC for inline cross section generation was accomplished
- Development of efficient algorithms for inline multigroup cross section generation is in progress