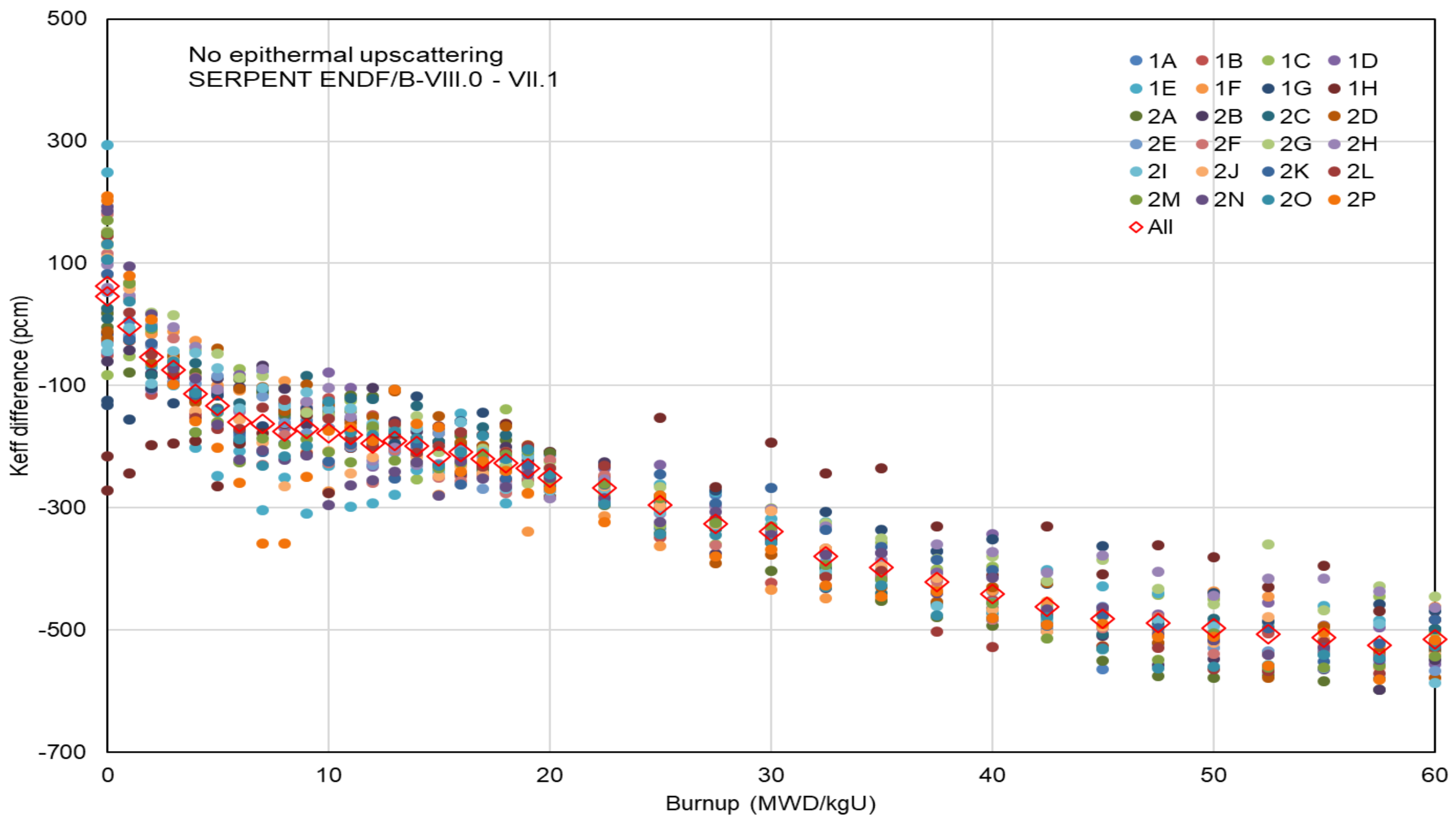
Investigation on the Reactivity Underestimation of ENDF/B-VIII.0 Compared to ENDF/B-VII.1 for Thermal Reactor Analysis

2020 CSEWG Meeting
December 2, 2020

SCALE XSProc Team

ENDF/B-VII.1 vs. ENDF/B-VIII.0 for Depletion

- **Reactivity underestimation**
  - VERA Depletion Benchmark Problems
    - PWR single pins and assemblies: SERPENT2 Monte Carlo
  - ENDF/B-VIII.0 reactivities are much lower
    - ^{235}U absorption cross section



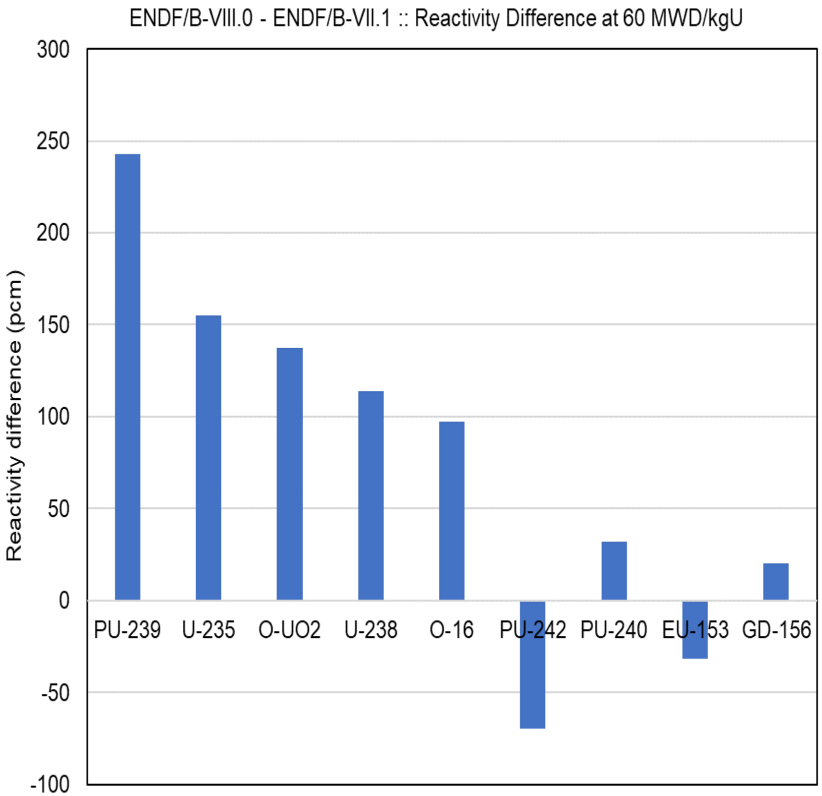
Detailed Investigation

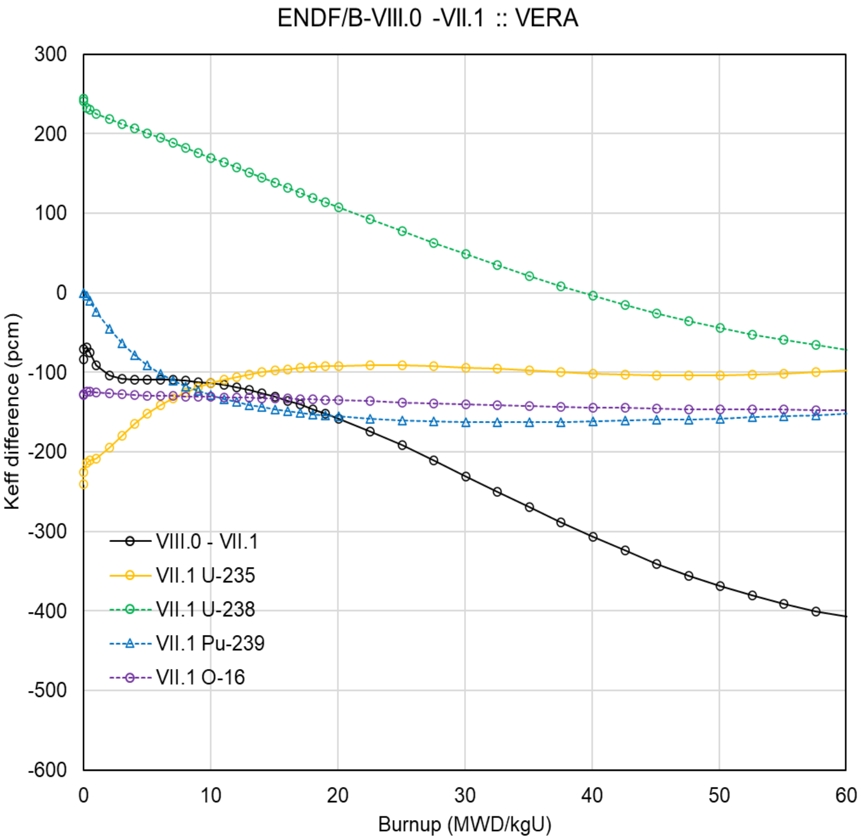
- **Neutronic simulators**
  - **VERA** : MPACT + SCALE-ORIGEN with 51-group
  - **Serpent-2** : Continuous energy Monte Carlo + depletion

- **Sensitivity calculations using VERA**
  - MPACT 51-group cross section library
    - ENDF/B-VII.1 and VIII.0 (reference)
    - 305 nuclides
    - Replace cross sections of VIII.0 for each nuclide with VII.1
    - Additional 305 sets of the MPACT 51-group libraries
  - Benchmark
    - Typical PWR single fuel pin
      - 3.1 w/o U-235
      - 900 K for fuel, 600 K for cladding and moderator
    - Depletion
      - ORIGEN-API with ORIGEN depletion libraries (255 burnup chains)
    - Result
      - Identify the most influencing nuclides
      - Verification calculation using Serpent only for the most influencing nuclides

Benchmark Results with VERA

- **Most influencing nuclides**
  - 60 MWD/kgU: Pu-239,U-235,O-16,U-238,Pu-242, Pu-240
  - Error cancellation: U-235+O-16 (negative) vs. U-238 (positive)
  - ρ difference (647 pcm) vs. Total summation of individual (654 pcm)



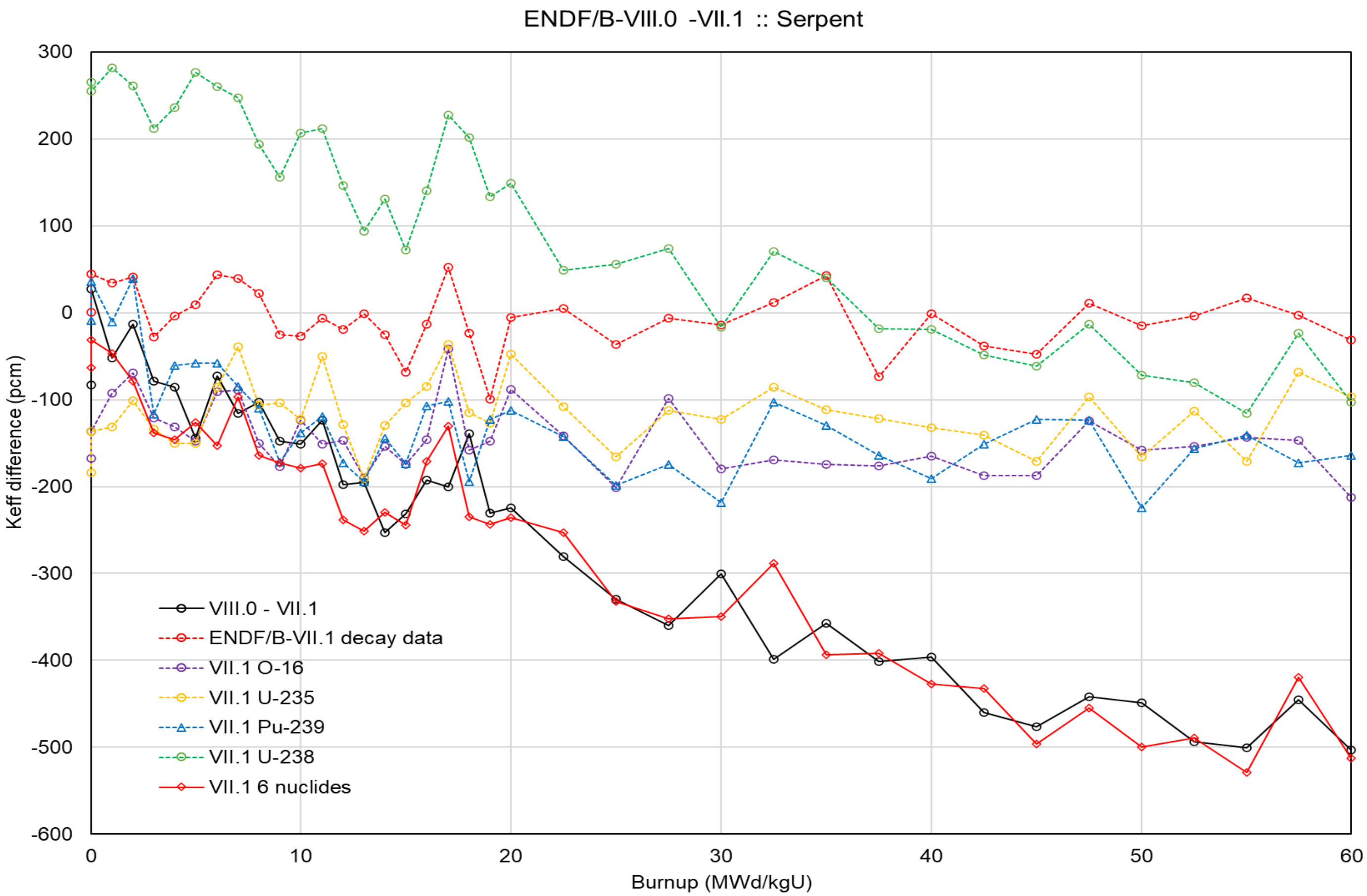


OAK RIDGE National Laboratory

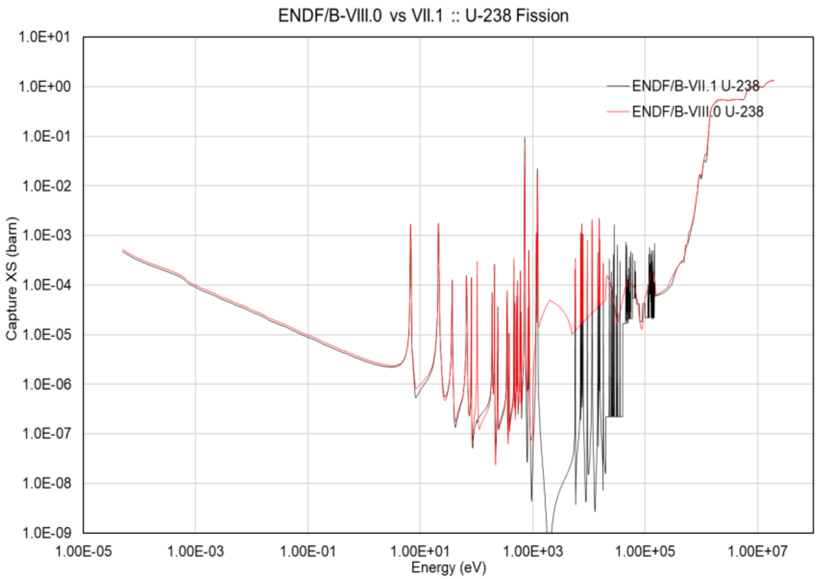
CASL The Consortium for Advanced Simulation of LWRs A DOE Energy Innovation Hub

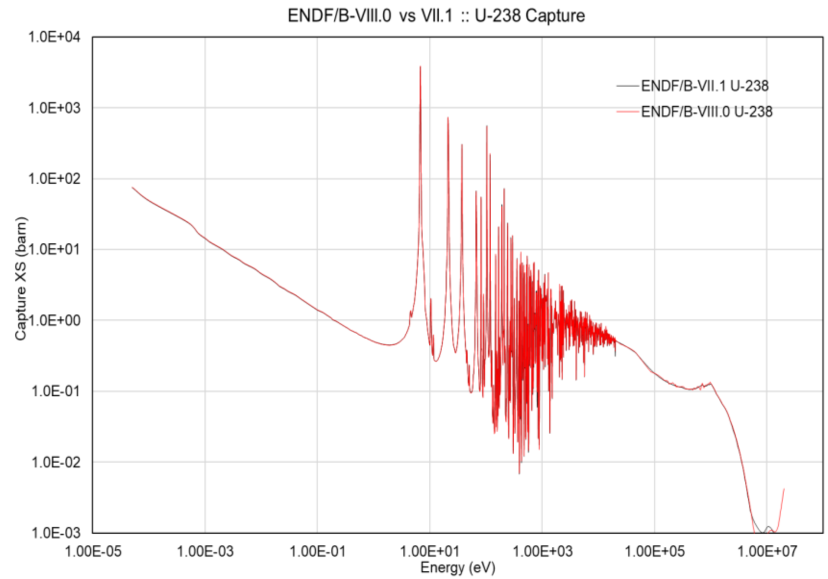
Benchmark Results with Serpent

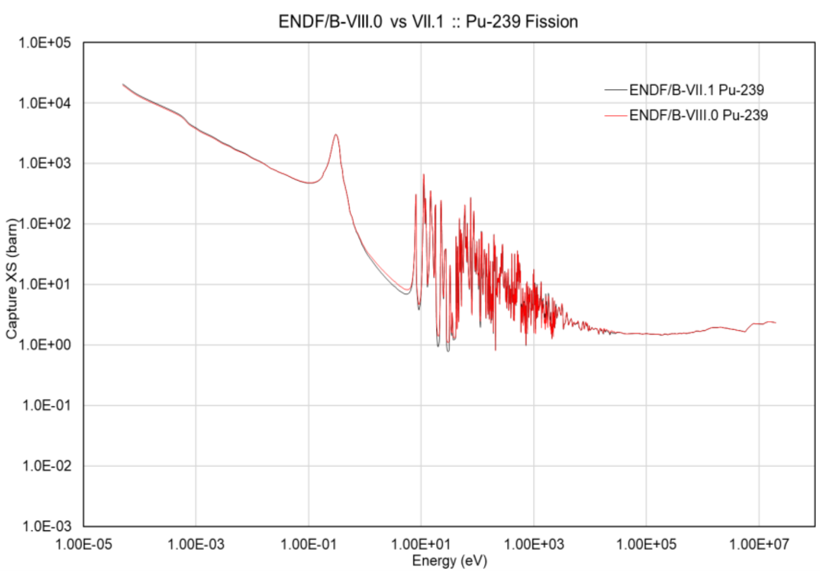
- **Most influencing nuclides**
  - Very consistent with the VERA results
  - With 6 ENF/B-VII.1 nuclides: same with ENDF/B-VII.1 result
  - ENDF/B-VII.1 decay and F.P. yield data: no impact

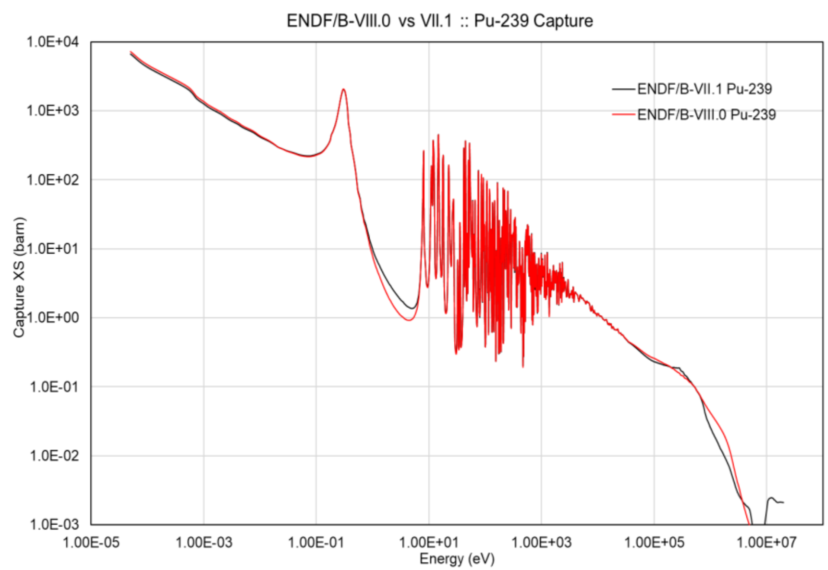


Direct Comparison of Cross sections I

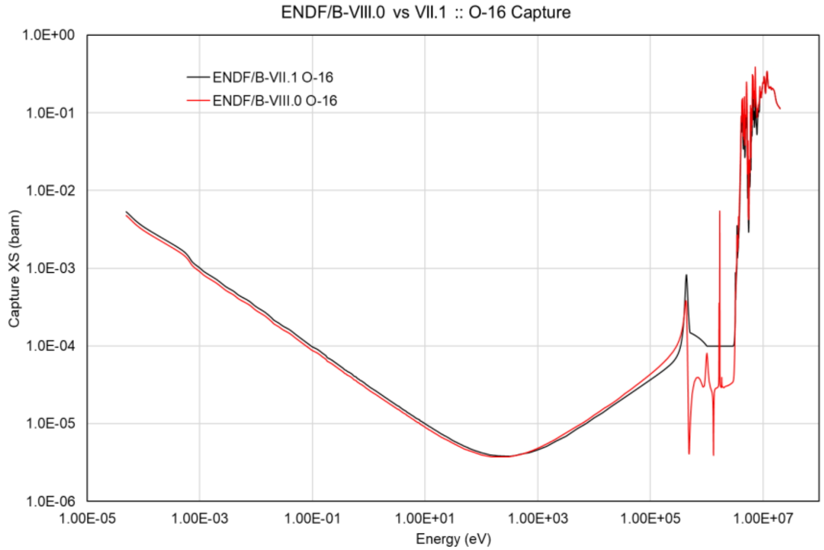




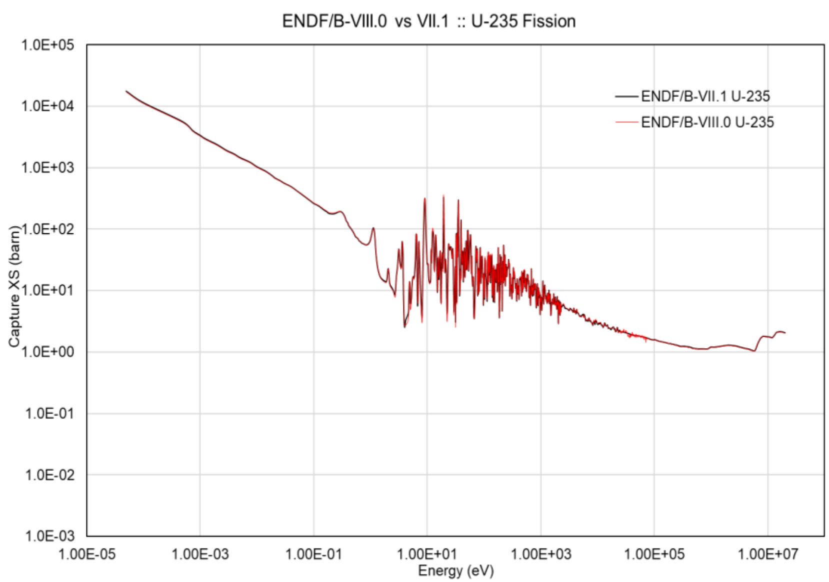


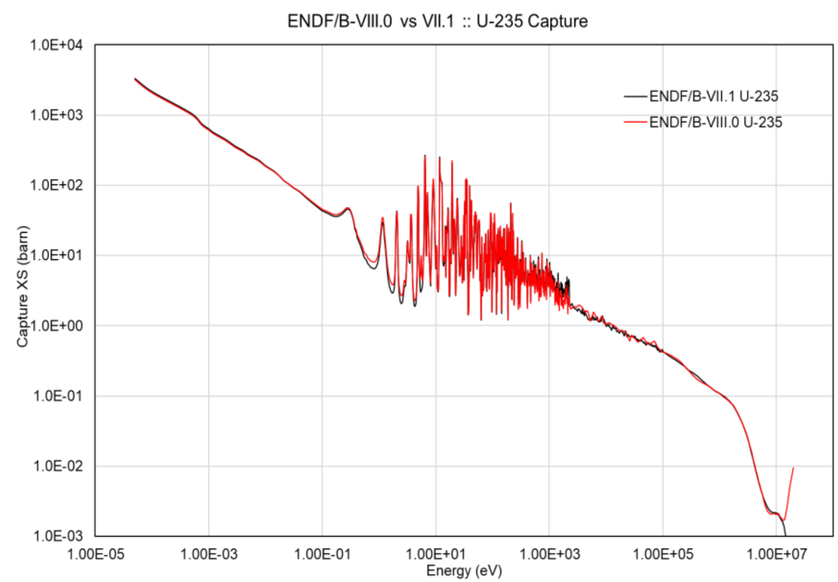


Direct Comparison of Cross sections II



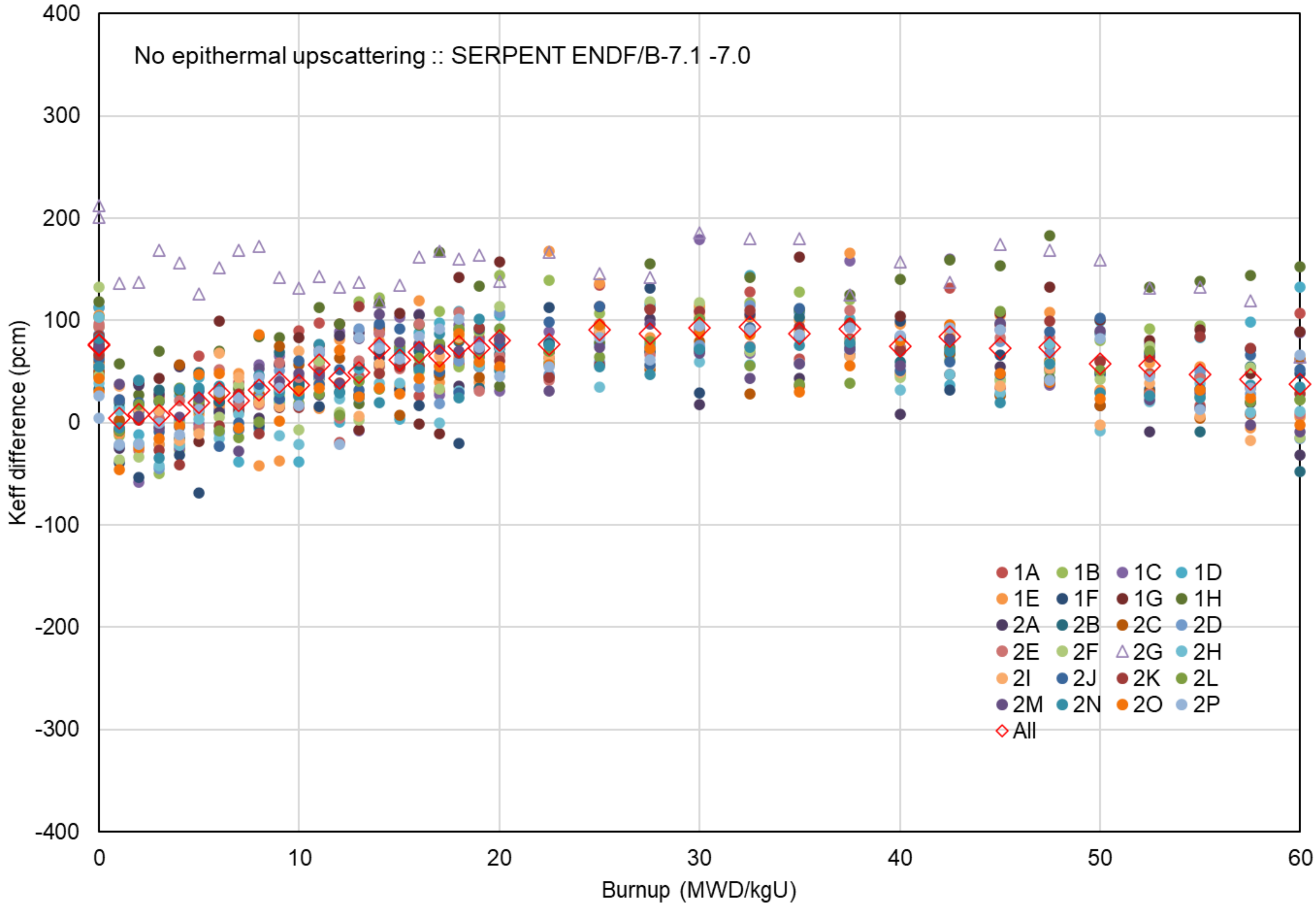
Case	Burnup	FPY	ENDF/B		Δρ (pcm)
			VII.1	VIII.0	
b01_burn_S00	0	no	1.24587	1.24720	86
b02_burn_S10	10	yes	1.08699	1.08738	33
b03_burn_S20	20	yes	1.00292	1.00297	5
b04_burn_S40	40	yes	0.88318	0.88297	-27
b05_burn_S60	60	yes	0.80886	0.80869	-26
b06_burn_S10x	10	no	1.17320	1.17394	54
b07_burn_S20x	20	no	1.11657	1.11647	-8
b08_burn_S40x	40	no	1.03682	1.03614	-63
b09_burn_S60x	60	no	0.98941	0.98849	-94





Discussion & Conclusion

- **ENDF/B-VII.1 vs. ENDF/B-VIII.0**
  - Most influencing nuclides
    - U-238, Pu-239, O-16 and U-235
    - U-238: +300 pcm at 0 burnup & getting decreased at high burnup
    - O-16: -150 pcm at all burnup steps
    - U-235: -150 pcm at all burnup steps
    - Pu-239: -200 pcm at high burnups
  - Error cancellation
    - U-238 (positive) vs. U-235 + O-16 (negative)
  - Decay data & F.P. yield data
    - No impact
  - Thermal reactor analysis
    - Generally accepted that even ENDF/B-VII.1 underestimates keff at high burnup
    - No epithermal upscattering
      - Considering epithermal upscattering would make it more negative
    - ENDF/B-VIII.0 may not be used for thermal reactor (PWR & BWR) analysis
  - ENDF/B release
    - May need to perform a sensitivity study for depletion effect


No epithermal upscattering :: SERPENT ENDF/B-7.1 -7.0
Keff difference (pcm)
400
300
200
100
0
-100
-200
-300
-400
0
10
20
30
40
50
60
Burnup (MWD/kgU)
1A 1B 1C 1D
1E 1F 1G 1H
2A 2B 2C 2D
2E 2F 2G 2H
2I 2J 2K 2L
2M 2N 2O 2P
All