Oak Ridge National Laboratory

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

2020 CSEWG Meeting
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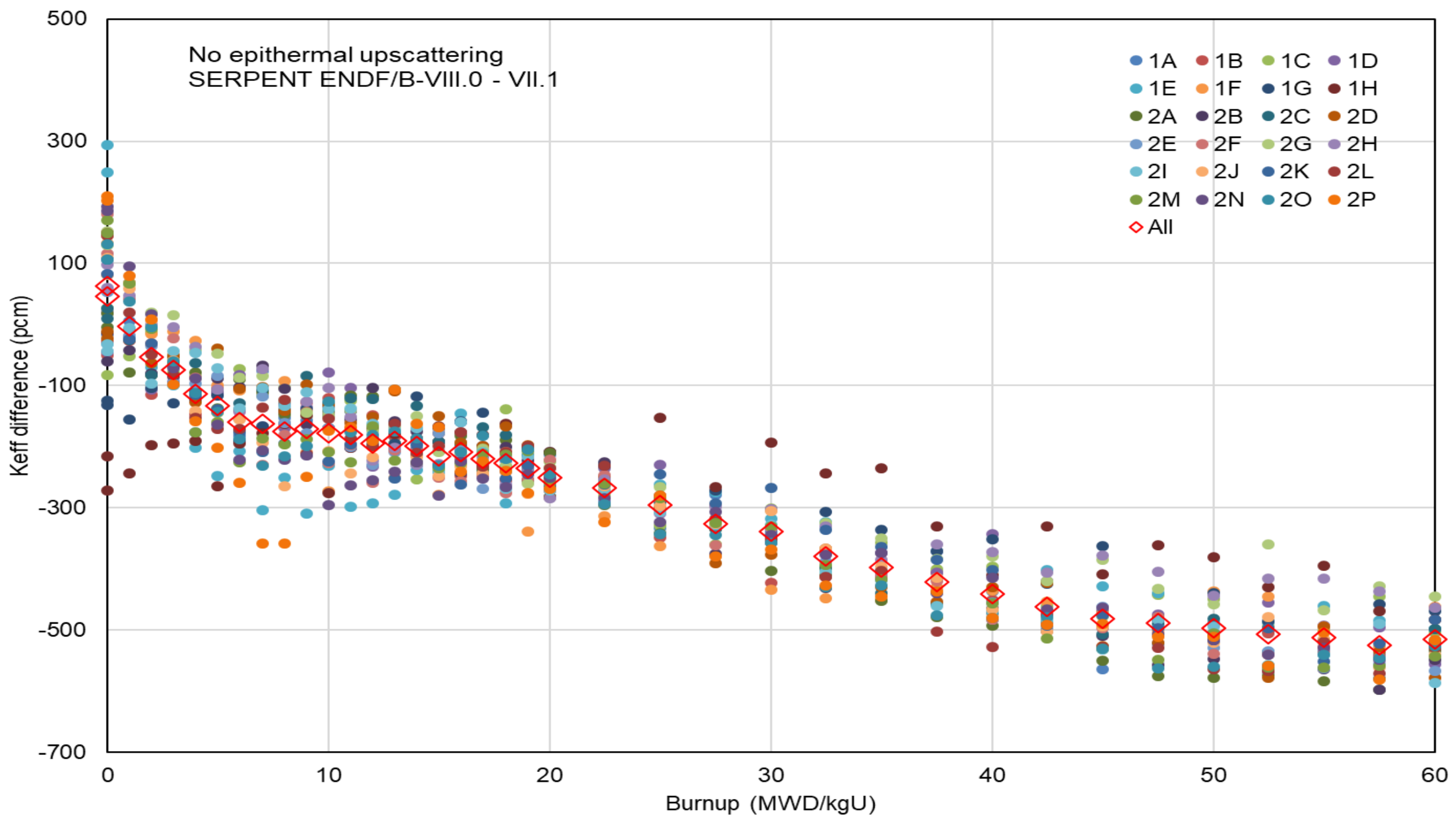
SCALE XSProc Team

ORNL is managed by UT-Battelle, LLC
for the US Department of Energy

U.S. DEPARTMENT OF ENERGY

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



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

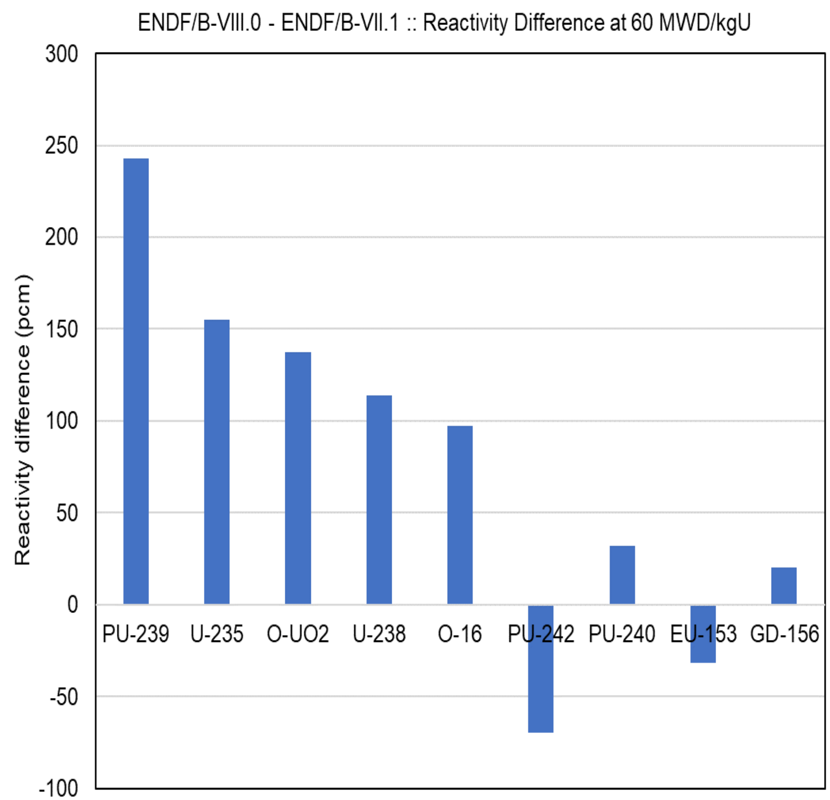
Detailed Investigation

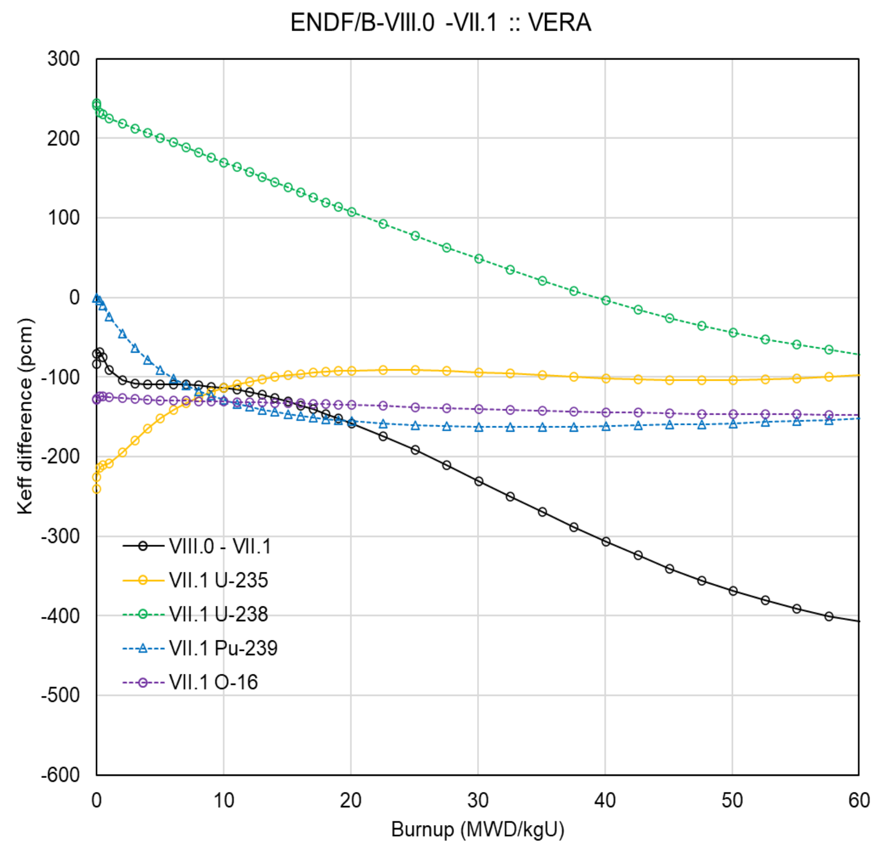
- **Neutronic simulators**
  - **VERA** : MPACT + SCALE-ORIGEN with 51-group
  - **Serpent-2** : Continuous energy Monte Carlo + depletion

- **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

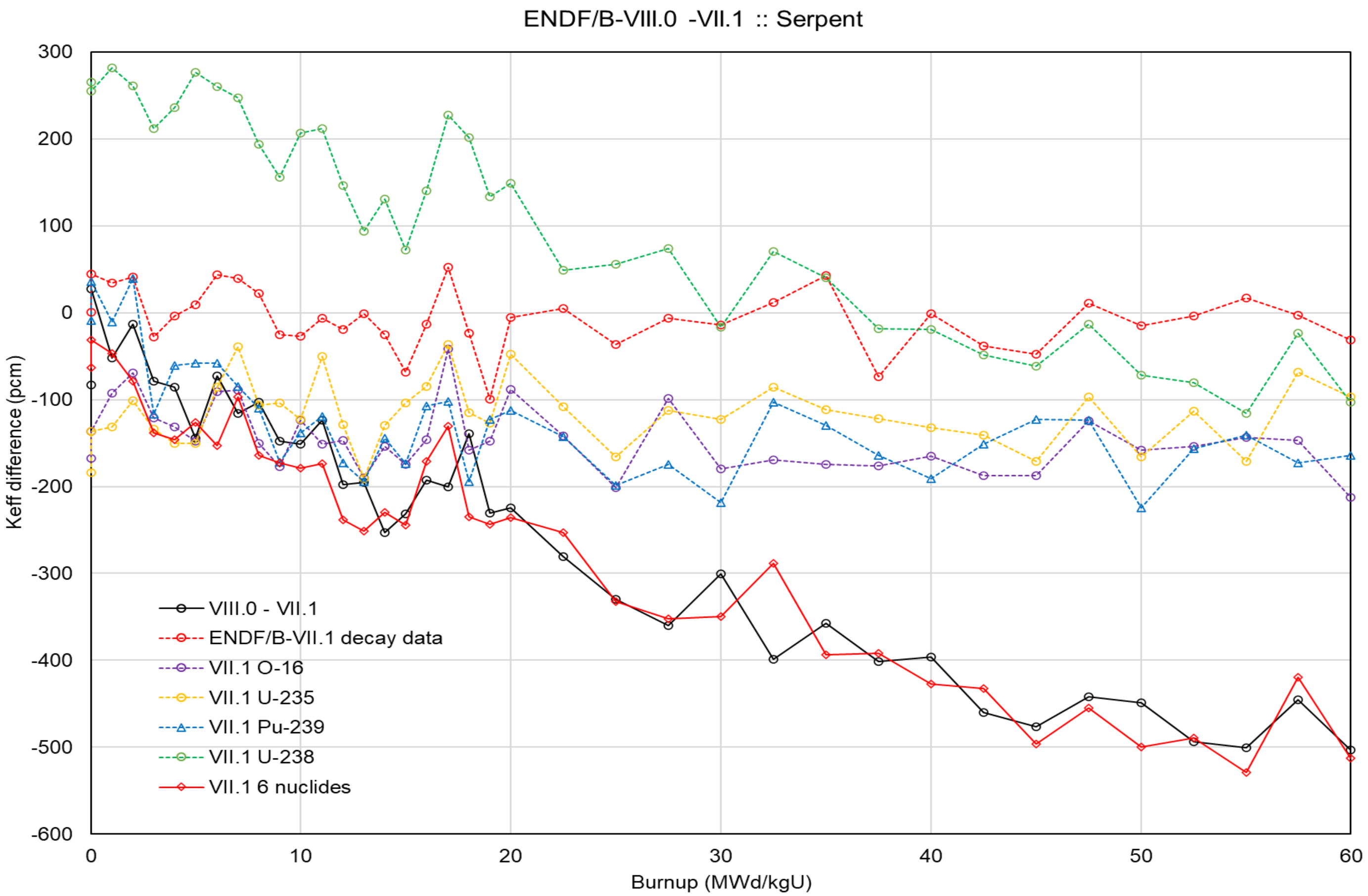
- **Most influencing nuclides**
  - 60 MWD/kgU: Pu-239,U-235,O-16,U-238,Pu-242, Pu-240
  - Error cancellation: U-235+O-16 (negative) vs. U-238 (positive)
  - ρ difference (647 pcm) vs. Total summation of individual (654 pcm)



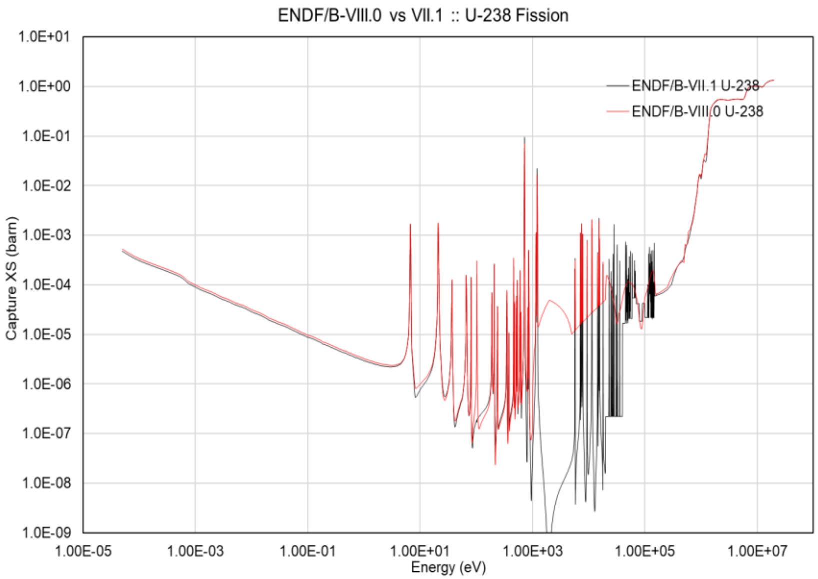


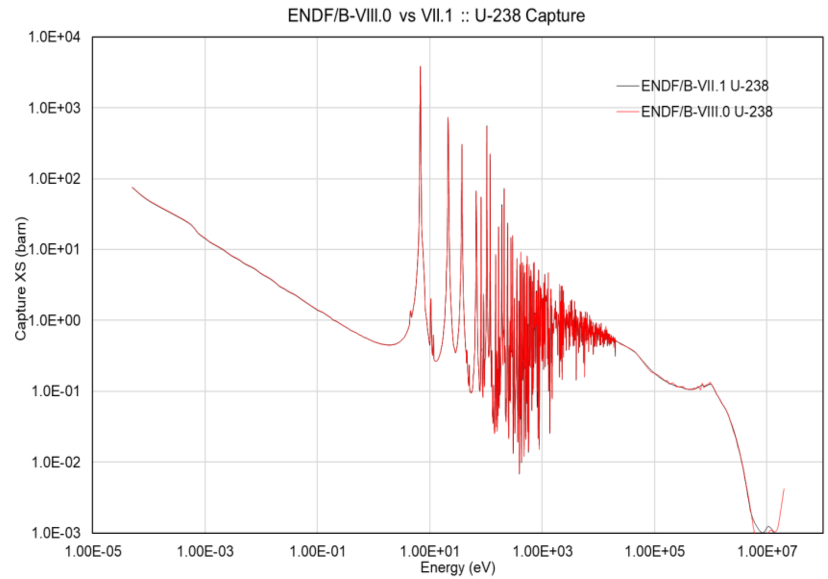
Benchmark Results with Serpent

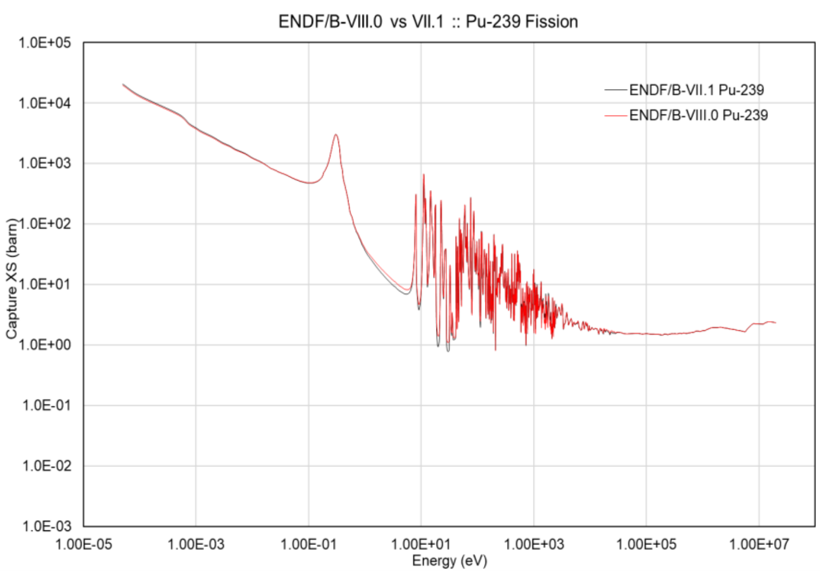
- **Most influencing nuclides**
  - Very consistent with the VERA results
  - With 6 ENF/B-VII.1 nuclides: same with ENDF/B-VII.1 result
  - ENDF/B-VII.1 decay and F.P. yield data: no impact

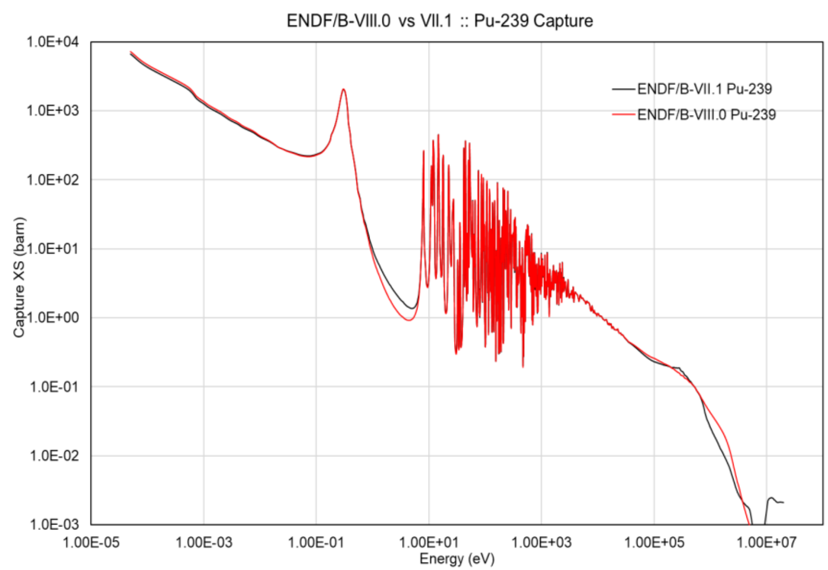


Direct Comparison of Cross sections I

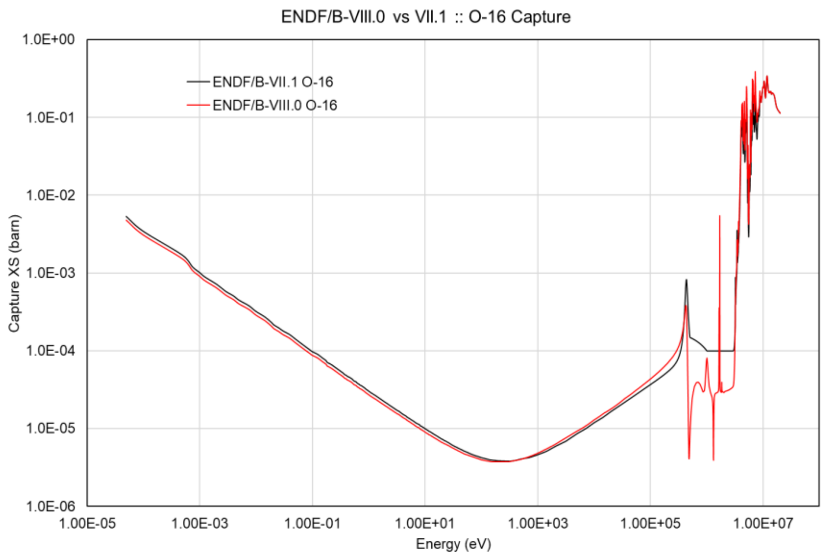




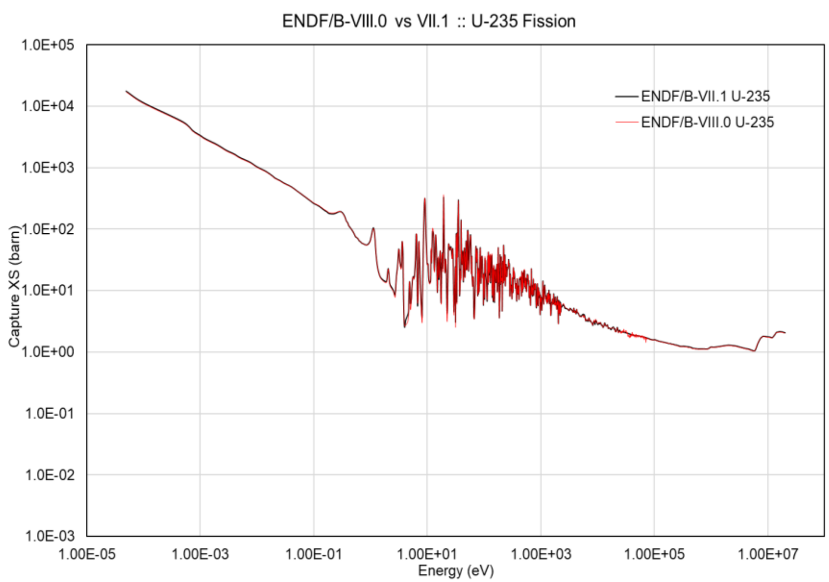


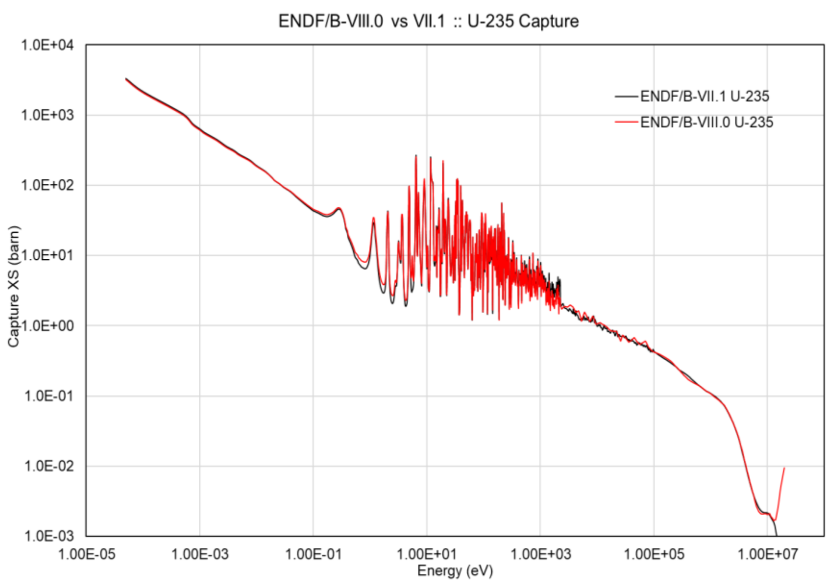


Direct Comparison of Cross sections II



Case	Burnup	FPY	ENDF/B VII.1	ENDF/B VIII.0	Δρ (pcm)
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