^{235}U Data Set Testing with (mostly) ICSBEP Benchmarks

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- Bettis has performed continuous energy Monte Carlo eigenvalue calculations for a variety of ICSBEP uranium fueled benchmarks with ENDF/B-VI.8 cross sections or ENDF/B-VI.8 plus a LANL/ORNL ^{235}U data set ("U235LA15B", internal date code is 7/29/2004).
  - Current results include benchmark k_{eff} normalization corrections from the 2004 ICSBEP Handbook.
- Eigenvalues presented on subsequent slides are calculated from tracking 50 million neutron histories in ten independent 5 million history jobs.
  - The 95% eigenvalue confidence interval, determined from the variance in the ten independent eigenvalue estimates, is typically less than 0.0005 Δk (i.e., roughly the size of the plot symbol).

- A majority of these calculations use models derived from ICSBEP evaluations:
  - xxx-SOL-THERM
    - 9 HEU evaluations, 31 critical configurations plus two ORNL experiments (L5, L6) not yet in ICSBEP.
      - 4 evaluations (8 critical configurations) include a H_2O reflector.
    - 9 LEU evaluations, 39 critical configurations
      - 4 evaluations (19 critical configurations) include a H_2O reflector.
  - HEU-MET-FAST-007 (ORNL)
  - HEU-MET-INTER-006 (LANL/ZEUS)
  - HEU-MET-FAST-072 (LANL/ZEUS)

Selected HEU-SOL-THERM ICSBEP Benchmarks

Benchmark	Description
HEU-SOL-THERM-001 (encompass “RF#”)	Unreflected cylinders (~28cm, ~33cm and ~51cm diameter) containing uranyl nitrate.
HEU-SOL-THERM-009 (Case 3 = ORNL L7)	Water reflected spheres (6.4-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-010	Water reflected sphere (9.7-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-011	Water reflected spheres (17-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-012 (ORNL L10)	Water reflected sphere (91-liter) containing uranium oxyfluoride.
HEU-SOL-THERM-013 (ORNL1)	Unreflected sphere (174-liter) containing uranium nitrate.
HEU-SOL-THERM-032 (ORNL10)	Unreflected sphere (48-inch diameter) containing uranyl nitrate.
HEU-SOL-THERM-042 (encompass ORNL12 through ORNL23)	Unreflected cylinders (~77cm and ~137cm diameter) containing uranium nitrate.
HEU-SOL-THERM-043 (Cases 2 and 3 are ORNL L8 and ORNL L9)	Unreflected spheres (17-liter, 91-liter and 174-liter) containing uranium oxyfluoride solution.
ORNL L5, L6	Unreflected cylinders (13cm diameter) of uranium oxyfluoride. Low $H/^{235}U$ ratios of 27 and 44 lead to large leakage.

Selected LEU-SOL-THERM ICSBEP Benchmarks

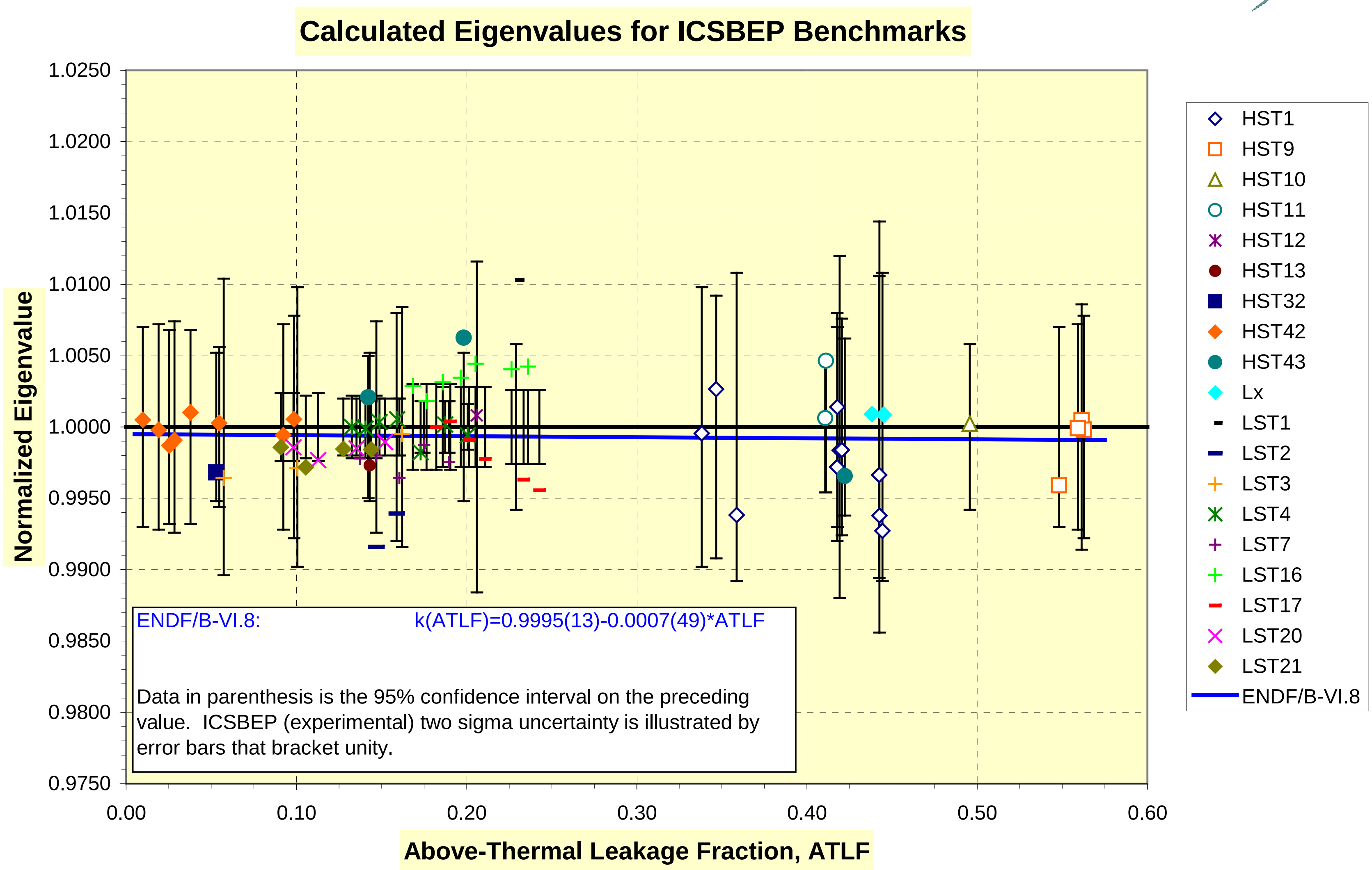
Benchmark	Description
LEU-SOL-THERM-001	"SHEBA-II". An unreflected UO_2F_2 + H_2O cylindrical (~20 inch diameter) assembly.
LEU-SOL-THERM-002	174-liter spheres of low enriched (4.9%) uranium oxyfluoride solutions. Case 1 is a water reflected solution, case 2 is unreflected.
LEU-SOL-THERM-003	Bare spheres of 10% enriched uranyl nitrate water solutions. Cases 3, 6 and 9 are full spheres with increasing H-to-^{235}U ratio.
LEU-SOL-THERM-004	STACY; Water reflected, 10% enriched uranyl nitrate solution in a 60cm diameter cylindrical tank. Seven cases with varying gU/liter.
LEU-SOL-THERM-007	STACY; Unreflected, 10% enriched uranyl nitrate solution in a 60cm diameter cylindrical tank. Five cases with varying gU/liter.
LEU-SOL-THERM-016	STACY; Water reflected, 10% enriched uranyl nitrate solution in a rectangular (slab, ~28cm x ~69cm) tank. Seven cases with varying gU/liter.
LEU-SOL-THERM-017	STACY; Unreflected, 10% enriched uranyl nitrate solution in a rectangular (slab, ~28cm x ~69cm) tank. Six cases with varying gU/liter.
LEU-SOL-THERM-020	STACY; Water reflected, 10% enriched uranyl nitrate solution in an 80cm diameter cylindrical tank. Four cases with varying gU/liter.
LEU-SOL-THERM-021	STACY; Unreflected, 10% enriched uranyl nitrate solution in an 80cm diameter cylindrical tank. Four cases with varying gU/liter.

HEU- and LEU-SOL-THERM Characteristics

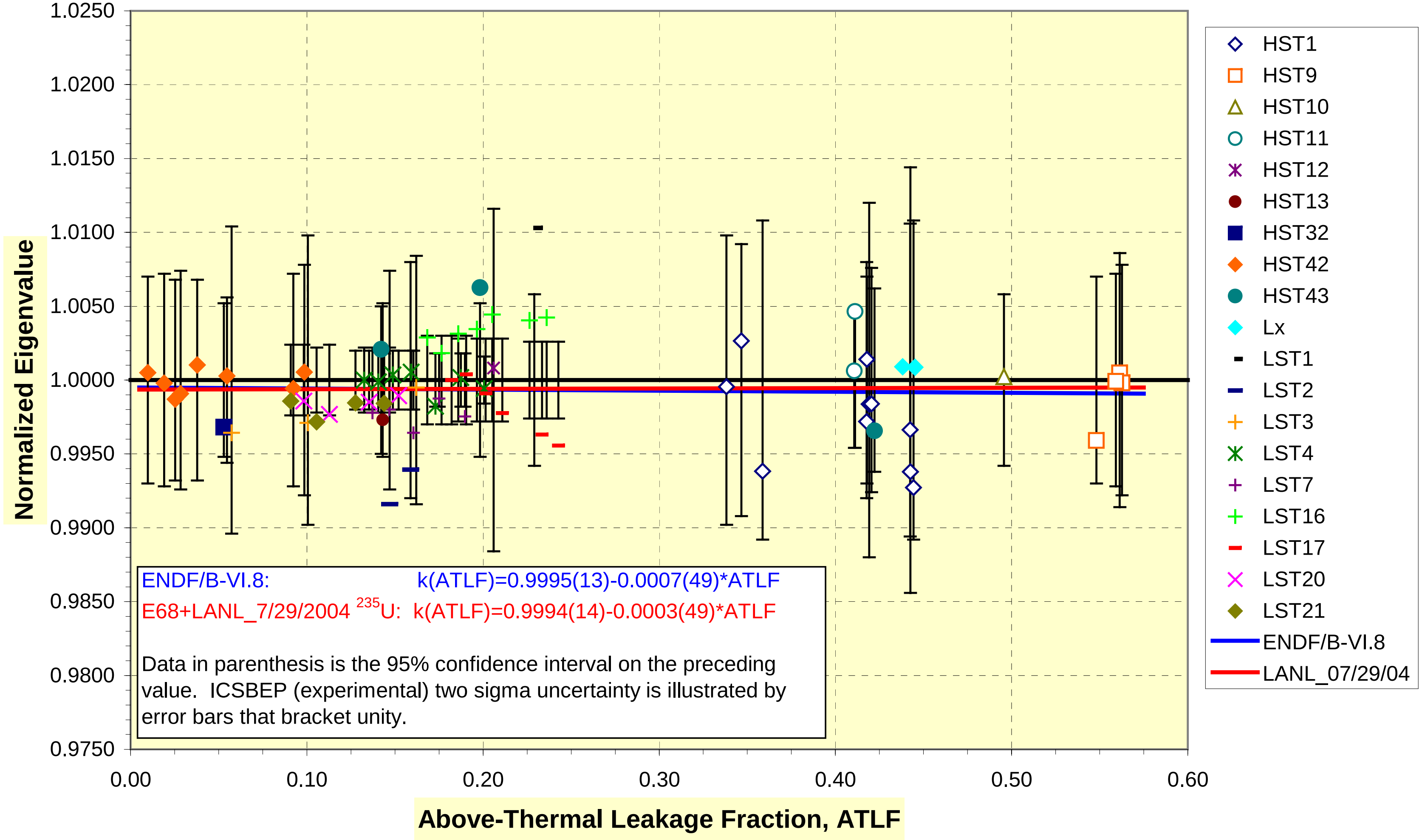
- Geometrically simple systems – spheres, cylinders or slabs with and without water reflection.
- Materially simple – $^{235,238}U$, hydrogen, oxygen plus small amounts of fluorine or nitrogen
  - Thin walled aluminum or stainless steel container is neutronically unimportant (but is still incorporated in modern models).
  - ^{235}U accounts for > 99% of fission.
  - ^{238}U absorption ranges from a fraction of a per cent in HEU solutions to several per cent in LEU STACY solutions.

HEU- and LEU-SOL-THERM Characteristics

- With ENDF/B-VI.8, these 33 HEU solution systems yield an average eigenvalue of 0.9993.
  - The 95% confidence interval for the average is ±0.0010 while the 95% confidence interval for the population is ±0.0059.
- With ENDF/B-VI.8, these 39 LEU solution systems yield an average eigenvalue of 0.9994.
  - The 95% confidence interval for the average is ±0.0011 while the 95% confidence interval for the population is ±0.0066.
- Conclude that HEU solution benchmarks and LEU solution benchmarks may be combined into a single database.


Calculated Eigenvalues for ICSBEP Benchmarks
Normalized Eigenvalue
1.0250
1.0200
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
0.9800
0.9750
0.00
0.10
0.20
0.30
0.40
0.50
0.60
Above-Thermal Leakage Fraction, ATLF
ENDF/B-VI.8:
k(ATLF)=0.9995(13)-0.0007(49)*ATLF
Data in parenthesis is the 95% confidence interval on the preceding value. ICSBEP (experimental) two sigma uncertainty is illustrated by error bars that bracket unity.
HST1
HST9
HST10
HST11
HST12
HST13
HST32
HST42
HST43
Lx
LST1
LST2
LST3
LST4
LST7
LST16
LST17
LST20
LST21
ENDF/B-VI.8

Calculated Eigenvalues for ICSBEP Benchmarks



Final xxx-SOL-THERM Observations

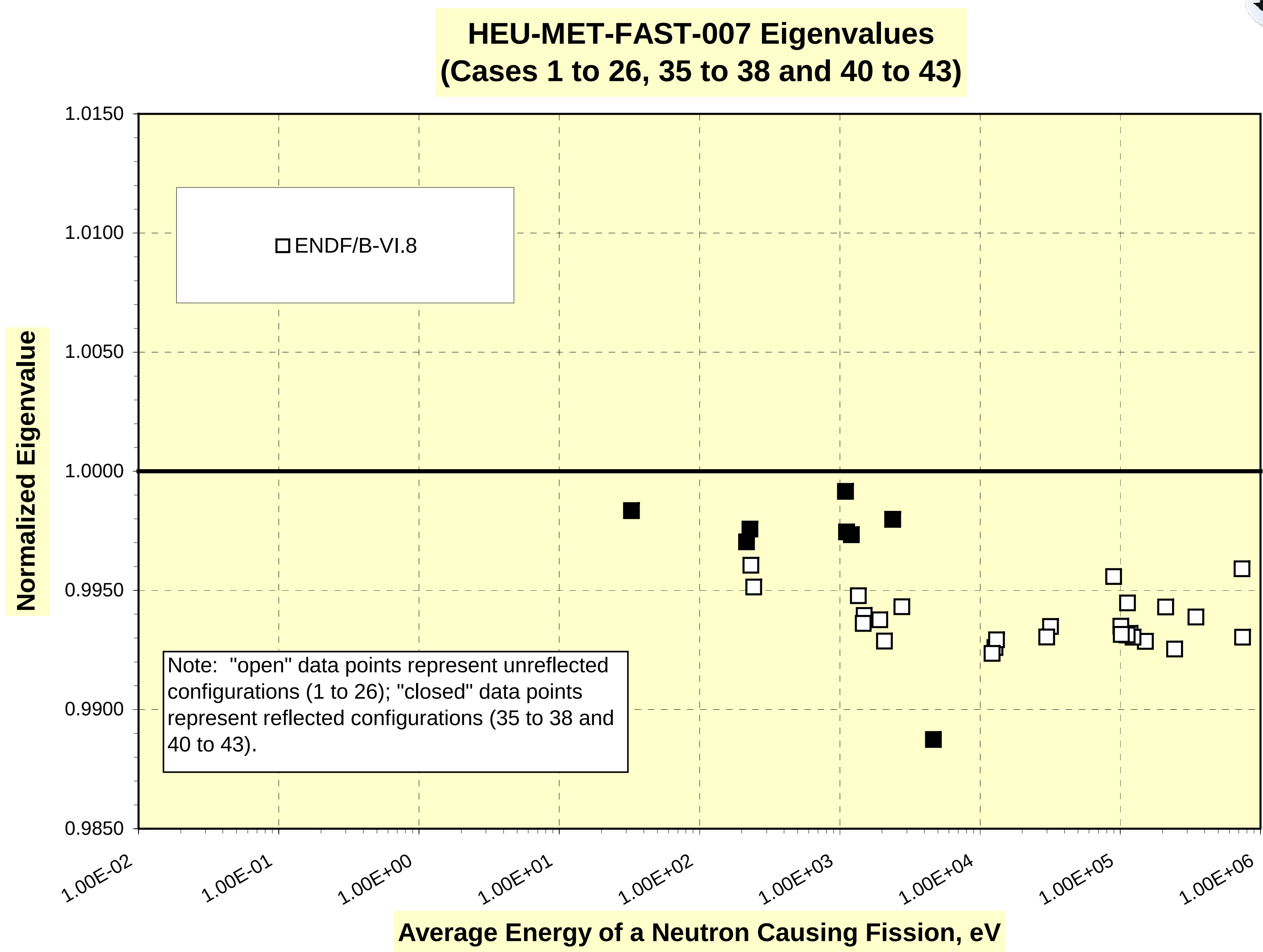
- Current ENDF/B-VI.8 (and JEFF-3.0 or JENDL-3.3) libraries yield accurate eigenvalues for ^{235}U solution systems.
  - HEU-SOL-THERM and LEU-SOL-THERM eigenvalues are virtually unity with no underlying eigenvalue trend as a function of Above-Thermal Leakage Fraction (ATLF).
- This excellent performance is retained with the LANL/ORNL U235LA15B (7/29/2004) data set.

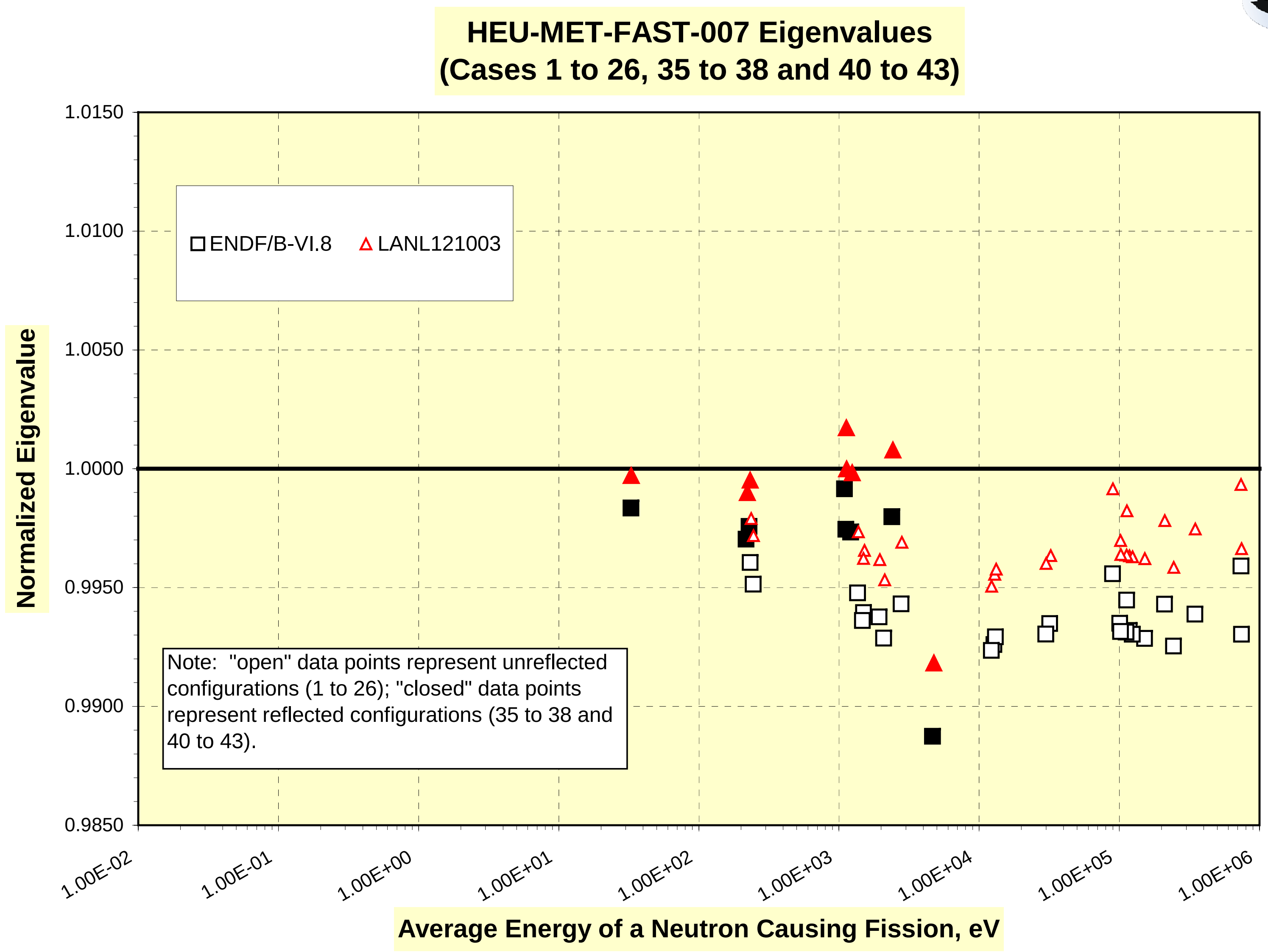
HEU-MET-xxx Testing

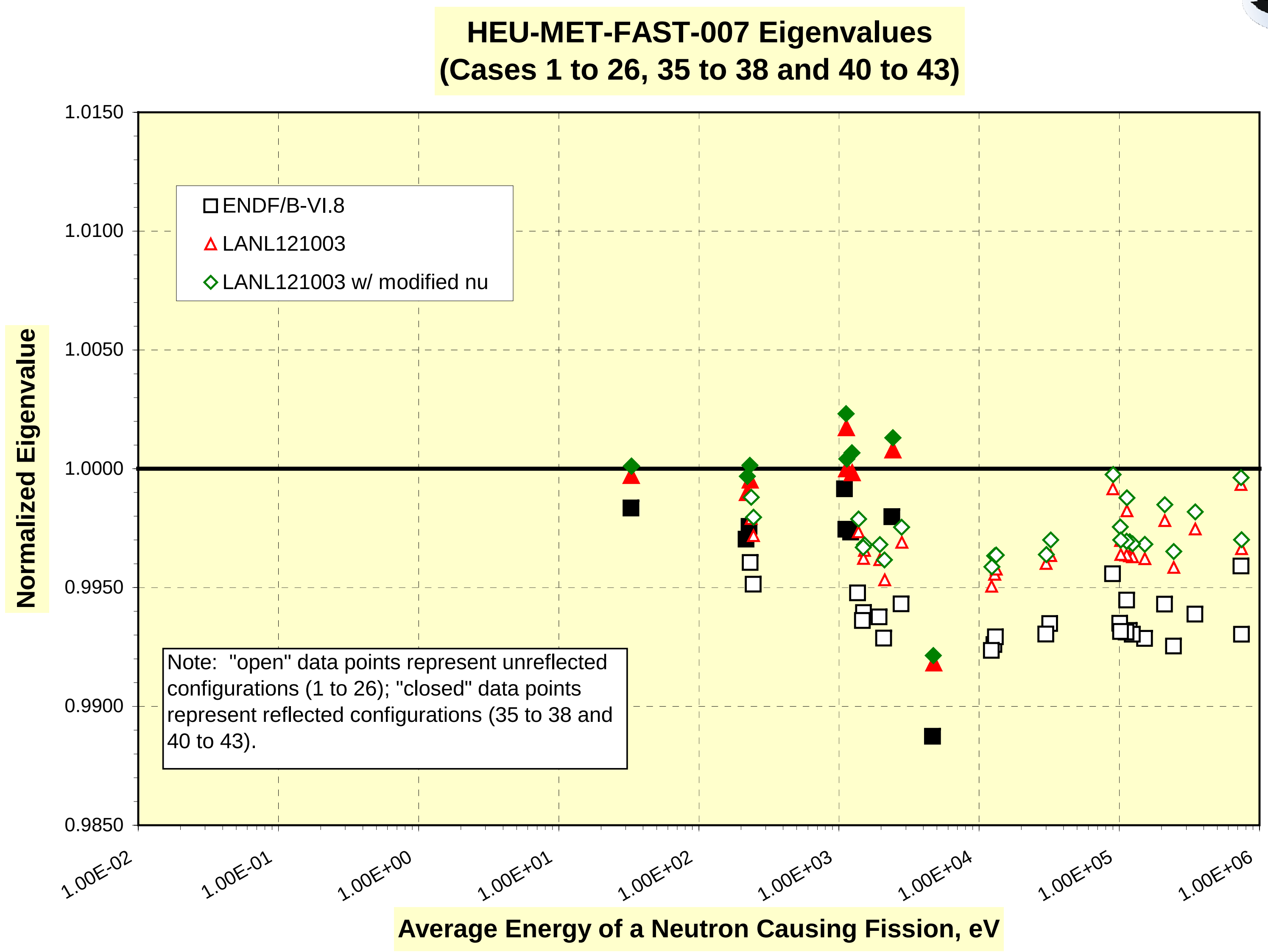
- HEU-MET-FAST-007 (ORNL)
  - 18 unreflected 10” x 10” and 8 unreflected 5” x 10” HEU slab configurations with interleaved sheets of polyethylene.
    - Average energy of a neutron causing fission is ~1.5 keV to ~750 keV.
  - 8 reflected 5” x 10” HEU slab configurations with interleaved polyethylene (reflector is 6” thick polyethylene on six sides).
    - Average energy of a neutron causing fission is ~30 eV to ~4.5 keV.

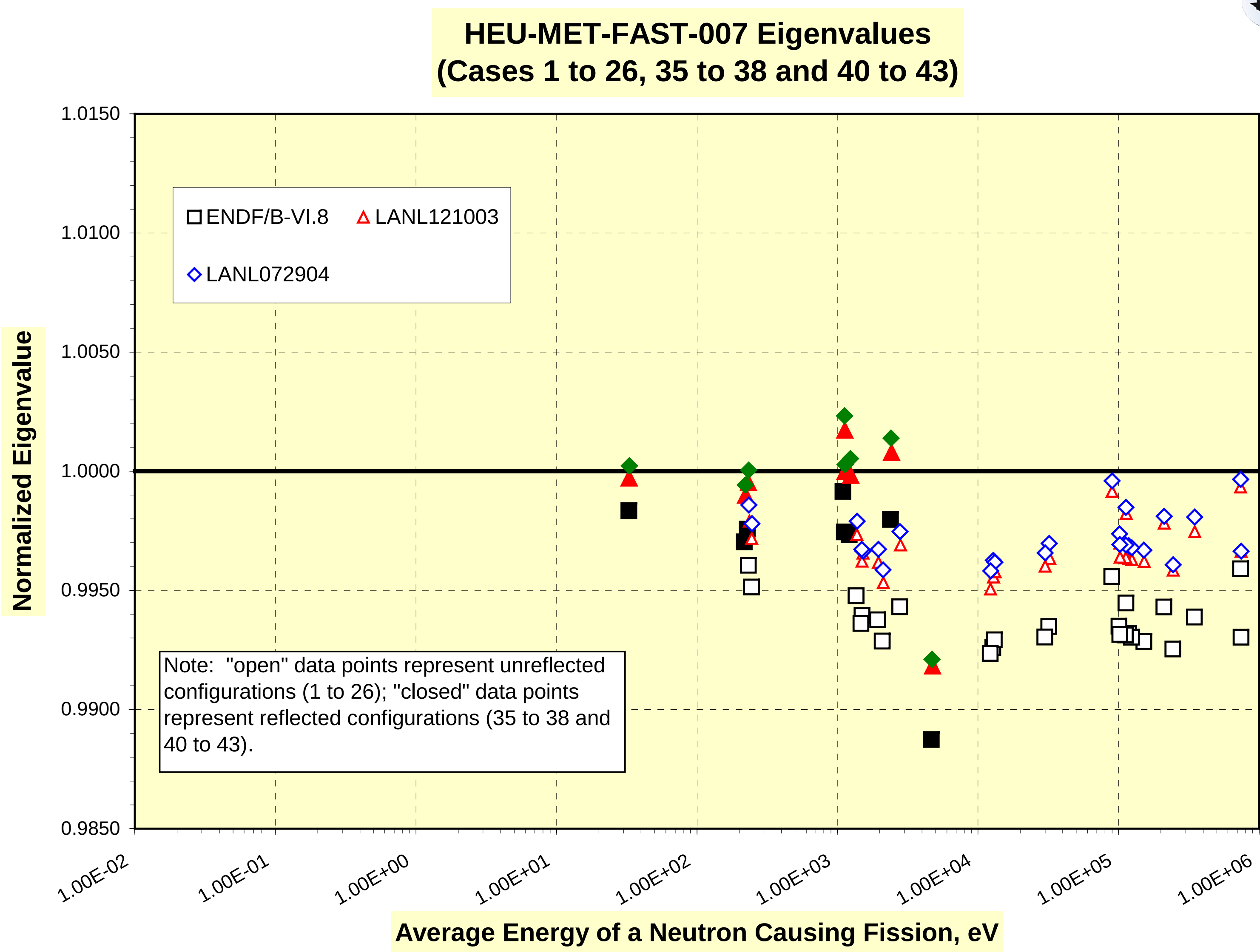
- HEU-MET-INTER-006 (LANL)
  - 4 copper reflected configurations. Repeating fuel unit is HEU and graphite.
    - Average energy of a neutron causing fission is ~3 keV to ~60 keV.

- HEU-MET-FAST-072 (LANL)
  - 2 copper reflected configurations. Repeating fuel unit is HEU and carbon steel.
    - Average energy of a neutron causing fission is ~190 keV.


HEU-MET-FAST-007 Eigenvalues
(Cases 1 to 26, 35 to 38 and 40 to 43)
Normalized Eigenvalue
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
ENDF/B-VI.8
Note: "open" data points represent unreflected configurations (1 to 26); "closed" data points represent reflected configurations (35 to 38 and 40 to 43).
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Average Energy of a Neutron Causing Fission, eV

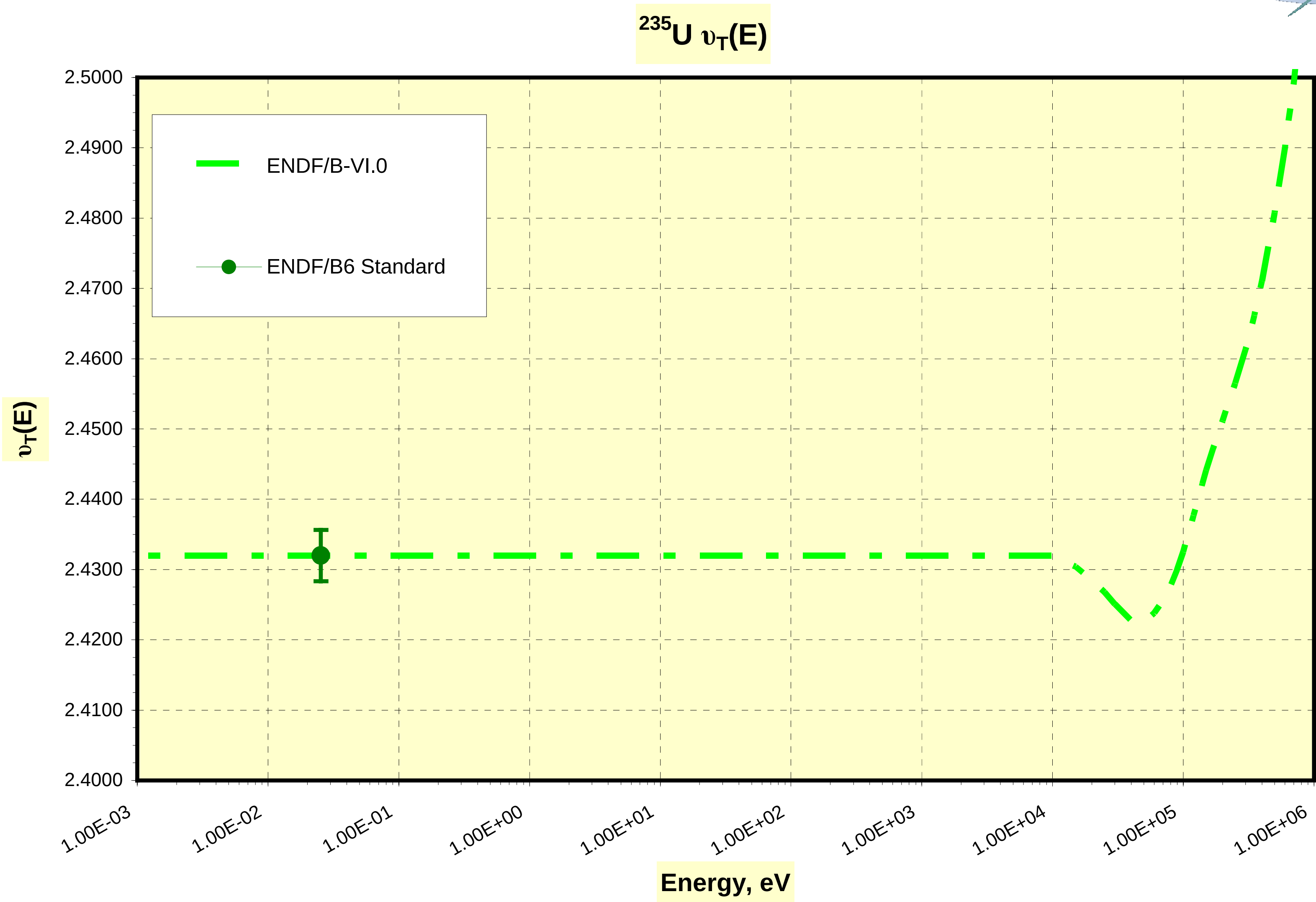

HEU-MET-FAST-007 Eigenvalues
(Cases 1 to 26, 35 to 38 and 40 to 43)
ENDF/B-VI.8
LANL121003
Note: "open" data points represent unreflected configurations (1 to 26); "closed" data points represent reflected configurations (35 to 38 and 40 to 43).
Normalized Eigenvalue
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Average Energy of a Neutron Causing Fission, eV


HEU-MET-FAST-007 Eigenvalues
(Cases 1 to 26, 35 to 38 and 40 to 43)
Normalized Eigenvalue
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
ENDF/B-VI.8
LANL121003
LANL121003 w/ modified nu
Note: "open" data points represent unreflected configurations (1 to 26); "closed" data points represent reflected configurations (35 to 38 and 40 to 43).
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Average Energy of a Neutron Causing Fission, eV

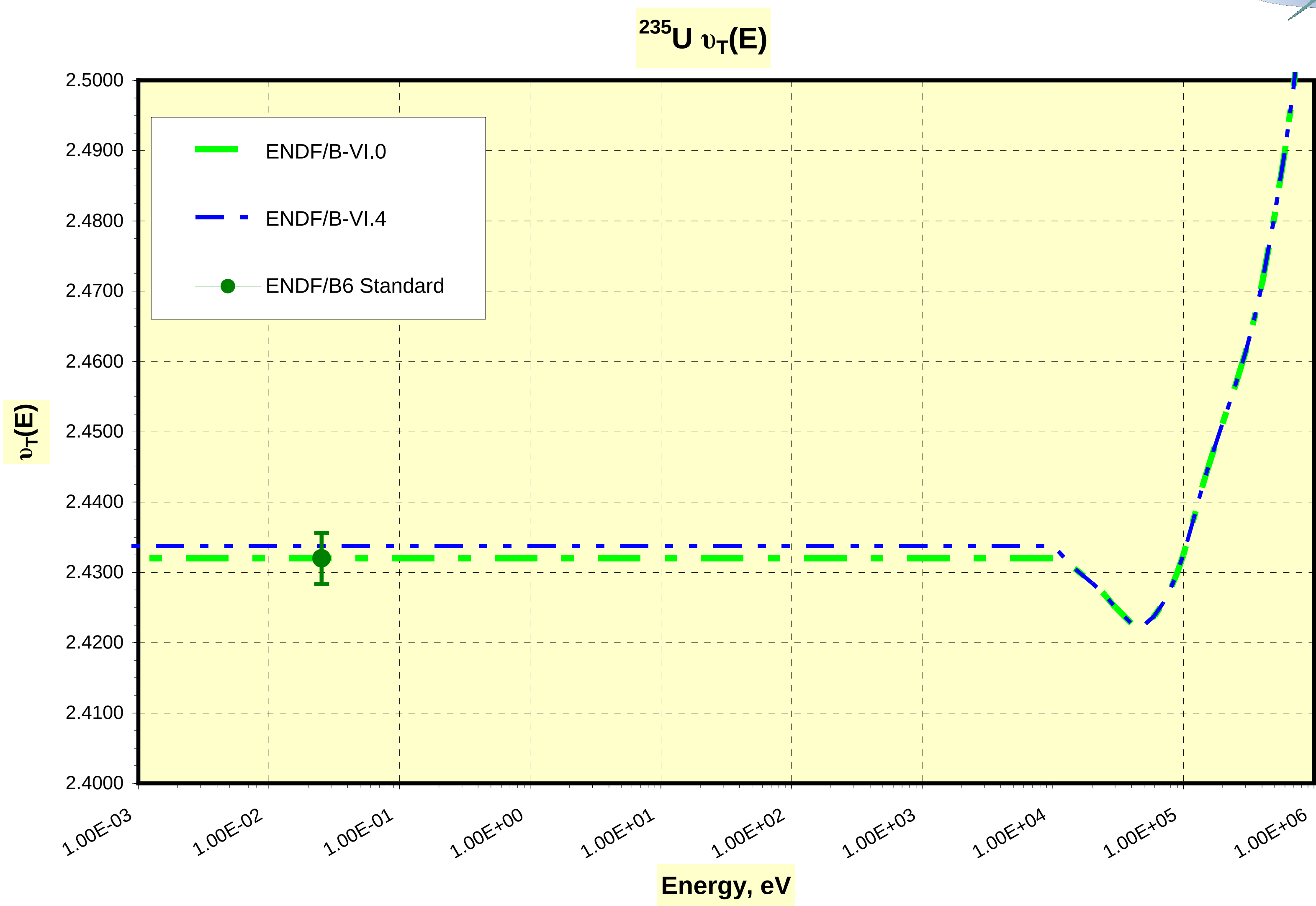

HEU-MET-FAST-007 Eigenvalues
(Cases 1 to 26, 35 to 38 and 40 to 43)
Normalized Eigenvalue
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
ENDF/B-VI.8
LANL121003
LANL072904
Note: "open" data points represent unreflected configurations (1 to 26); "closed" data points represent reflected configurations (35 to 38 and 40 to 43).
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Average Energy of a Neutron Causing Fission, eV

HEU-MET-xxx Testing

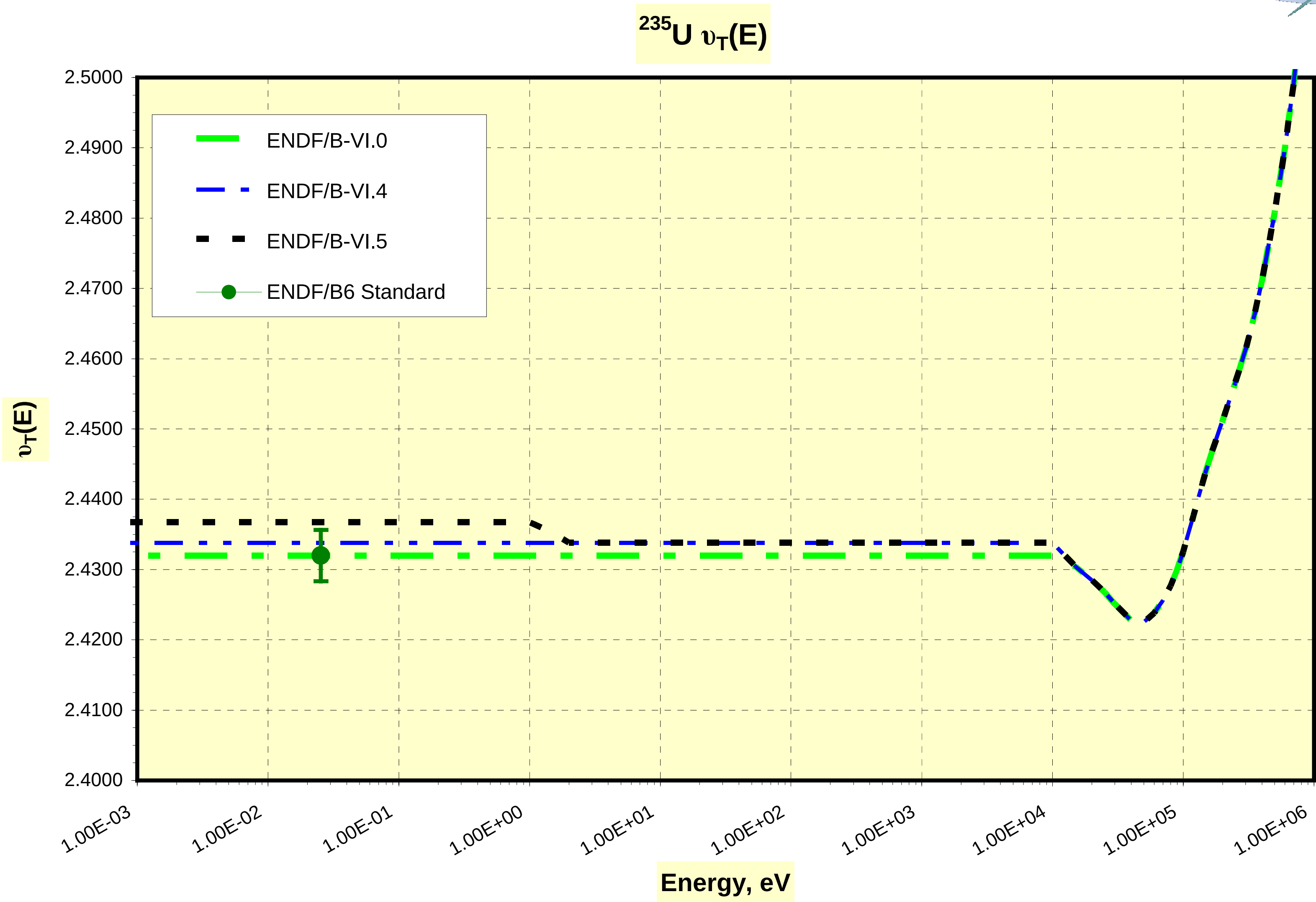
- HEU-MET-FAST-007 (ORNL)
  - A ~0.3% eigenvalue bias is observed between Reflected and Unreflected Configurations, independent of ^{235}U cross section data set.
  - There is no trend in calculated eigenvalue as a function of the Average Energy of a Neutron Causing Fission.
  - There is a pronounced dip in all ENDF/B-VI ^{235}U $\nu(E)$ evaluations between ~10 keV and ~100 keV. This dip was not present in the ENDF/B-V evaluation, but is not responsible for the reflector eigenvalue bias (nor, as will be shown, does it reduce the eigenvalue trend observed in the LANL/ZEUS benchmark analyses).



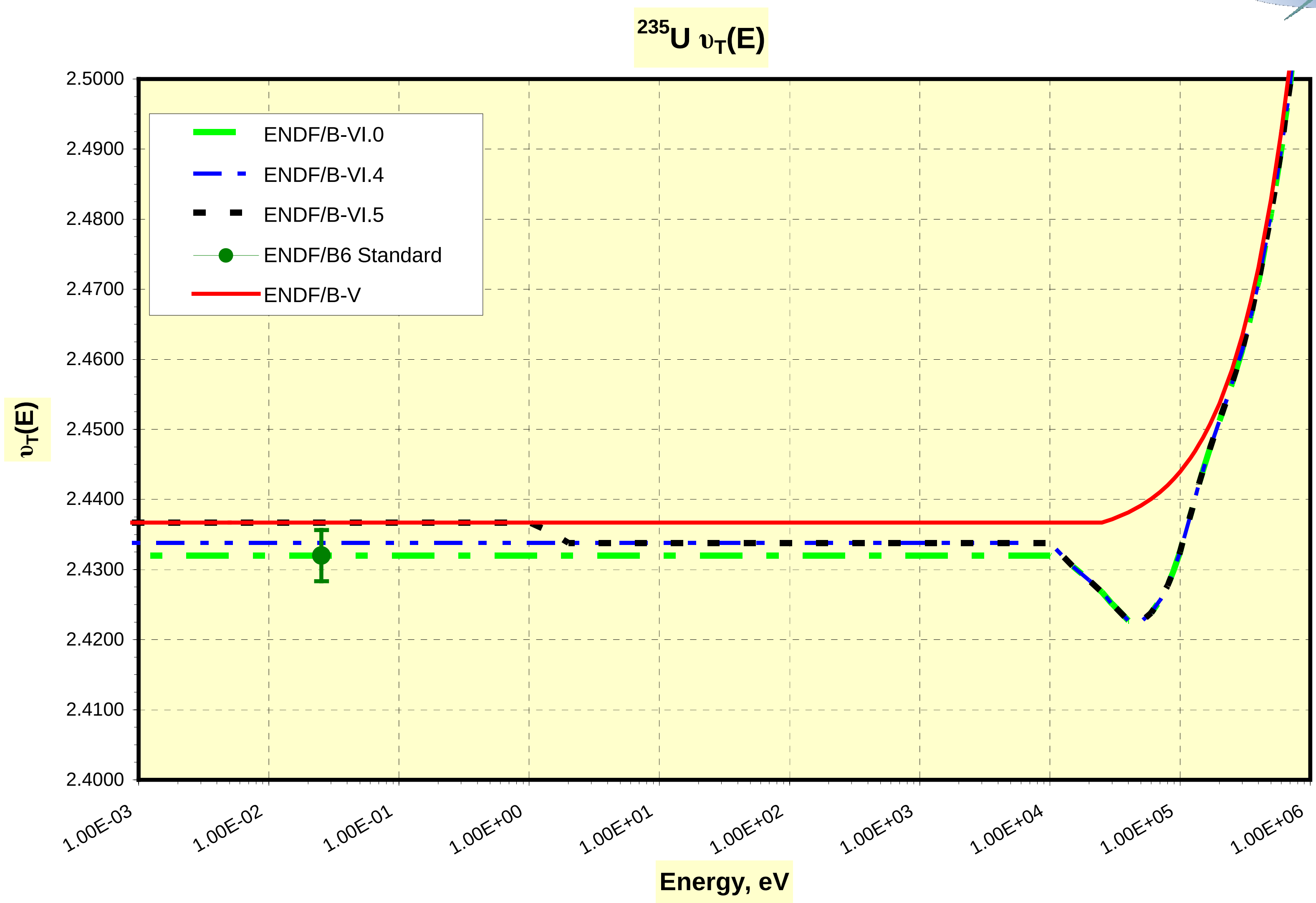
^{235}U υ_T(E)
2.5000
2.4900
2.4800
2.4700
2.4600
2.4500
2.4400
2.4300
2.4200
2.4100
2.4000
ENDF/B-VI.0
ENDF/B6 Standard
υ_T(E)
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Energy, eV



^{235}U υ_T(E)
ENDF/B-VI.0
ENDF/B-VI.4
ENDF/B6 Standard
2.5000
2.4900
2.4800
2.4700
2.4600
2.4500
2.4400
2.4300
2.4200
2.4100
2.4000
υ_T(E)
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Energy, eV



^{235}U υ_T(E)
2.5000
2.4900
2.4800
2.4700
2.4600
2.4500
2.4400
2.4300
2.4200
2.4100
2.4000
υ_T(E)
ENDF/B-VI.0
ENDF/B-VI.4
ENDF/B-VI.5
ENDF/B6 Standard
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Energy, eV



$^{235}U\ \nu_T(E)$
2.5000
2.4900
2.4800
2.4700
2.4600
2.4500
2.4400
2.4300
2.4200
2.4100
2.4000
$\nu_T(E)$
ENDF/B-VI.0
ENDF/B-VI.4
ENDF/B-VI.5
ENDF/B6 Standard
ENDF/B-V
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
Energy, eV

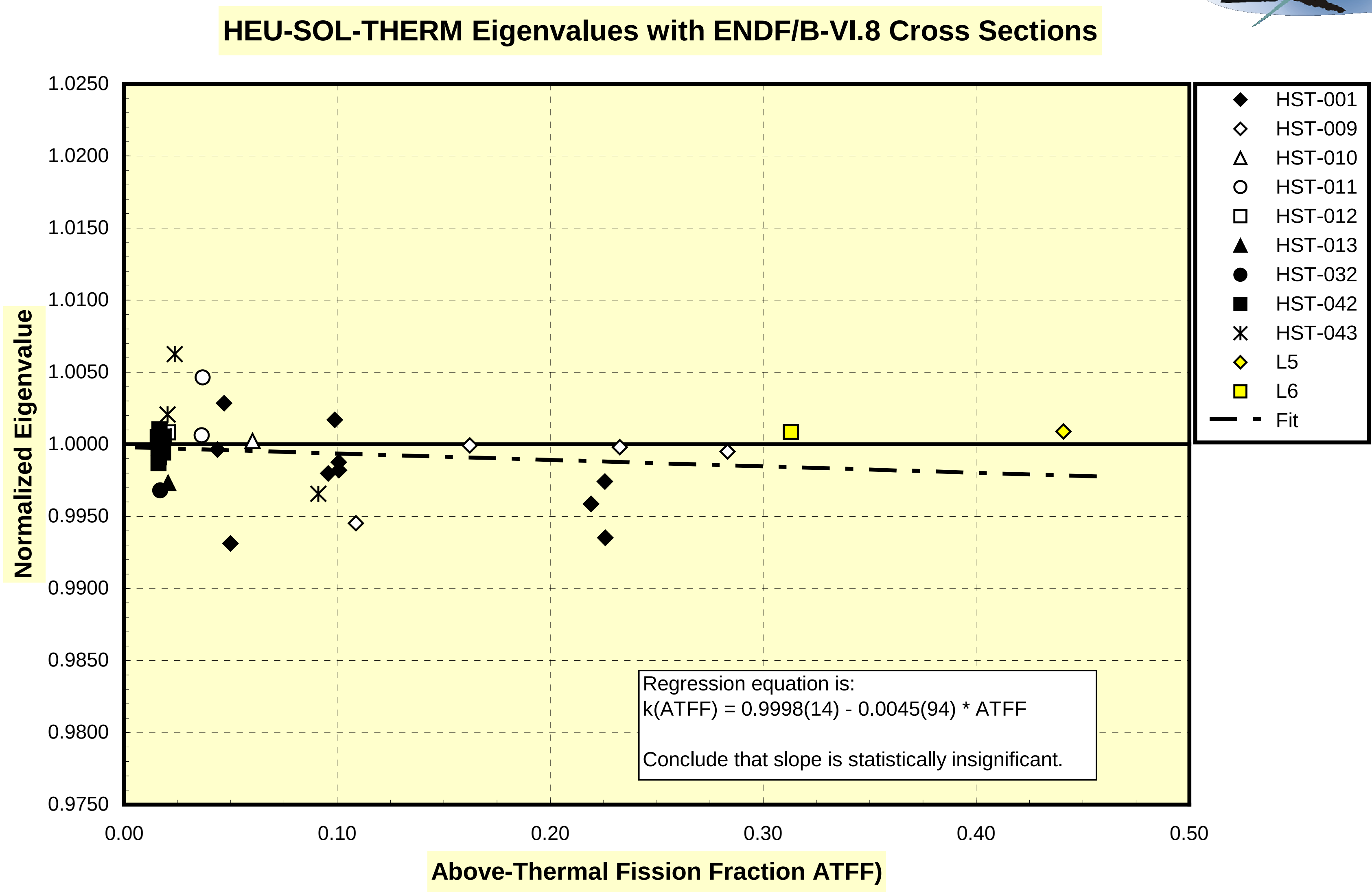
HEU-MET-INTER-006 and HEU-MET-FAST-072 Eigenvalues
(ZEUS Experiments with HEU and graphite or iron plus a copper reflector)

Benchmark	Average Energy of a Neutron Causing Fission, keV	ICSBEP ENDF/B-VI	Bettis ENDF/B-VI.8	Bettis ENDF/B-VI.8 plus LANL 12/10/2003 ^{235}U	Bettis ENDF/B-VI.8 plus LANL 12/10/2003 ^{235}U and modified $\nu(E)$	Bettis ENDF/B-VI.8 plus LANL 7/29/2004 ^{235}U
HMI-6.1	3.1	0.9912(6) [1]	0.98629(25)	0.98824(45)	0.98961(27)	0.98891(35)
HMI-6.2	6.5	0.9956(6)	0.99024(39)	0.99235(31)	0.99375(21)	0.99229(32)
HMI-6.3	15.7	0.9987(6)	0.99265(52)	0.99520(40)	0.99670(24)	0.99546(28)
HMI-6.4	58.9	1.0049(6)	0.99861(24)	1.00168(39)	1.00328(37)	1.00174(30)
HMF-72.1	190.9	1.0069(6) [2]	1.00230(24)	1.00452(35)	1.00569(25)	1.00432(31)
HMF-72.2	192.4	1.0073(6)	1.00346(35)	1.00550(20)	1.00681(32)	1.00517(38)

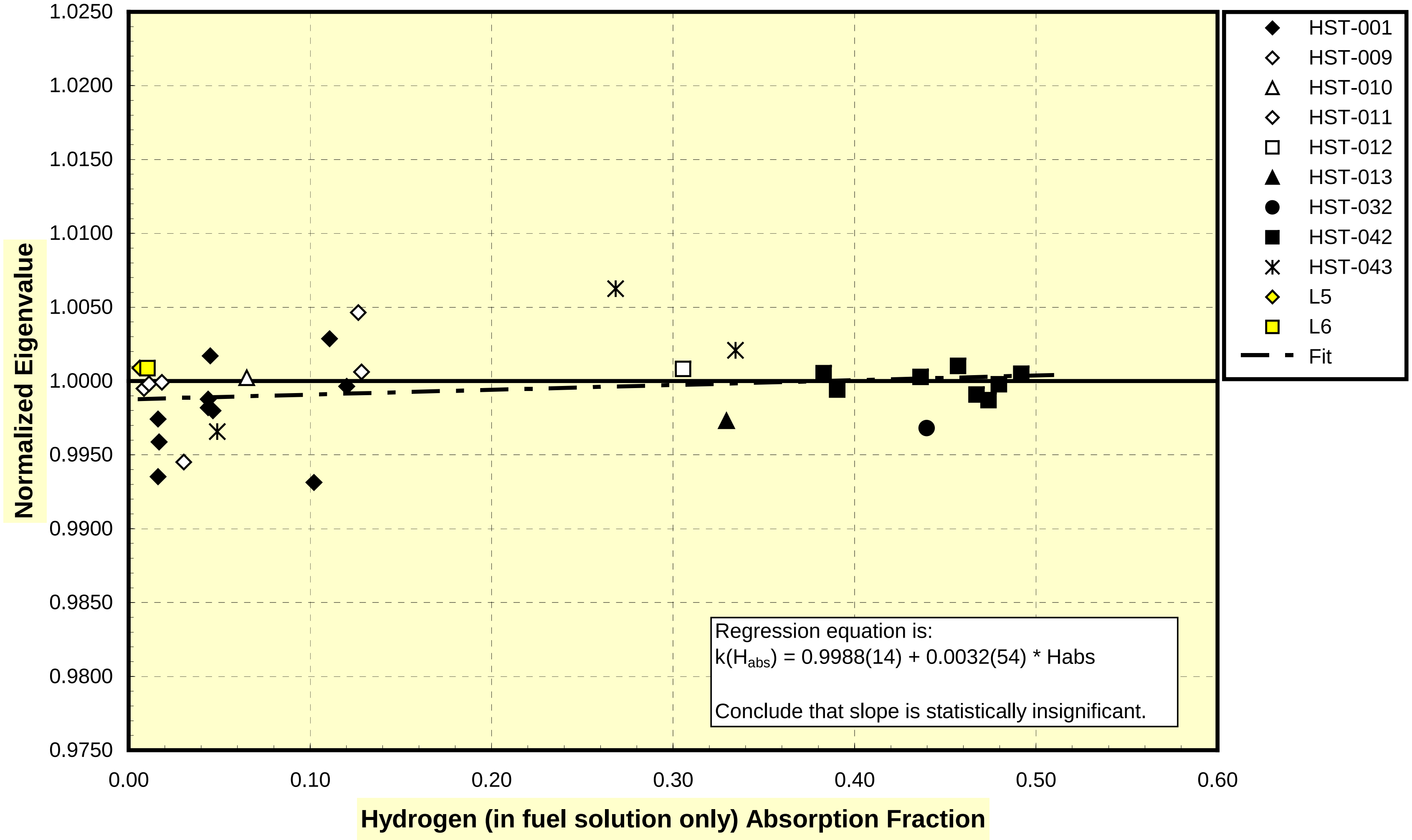
[1] MCNP listing in HEU-MET-INTER-006, Appendix A, shows use of ".66c" (i.e., ENDF/B-VI.8) uranium cross sections.
[2] HEU-MET-FAST-072, Table 40, results with ENDF/B-VI.4.

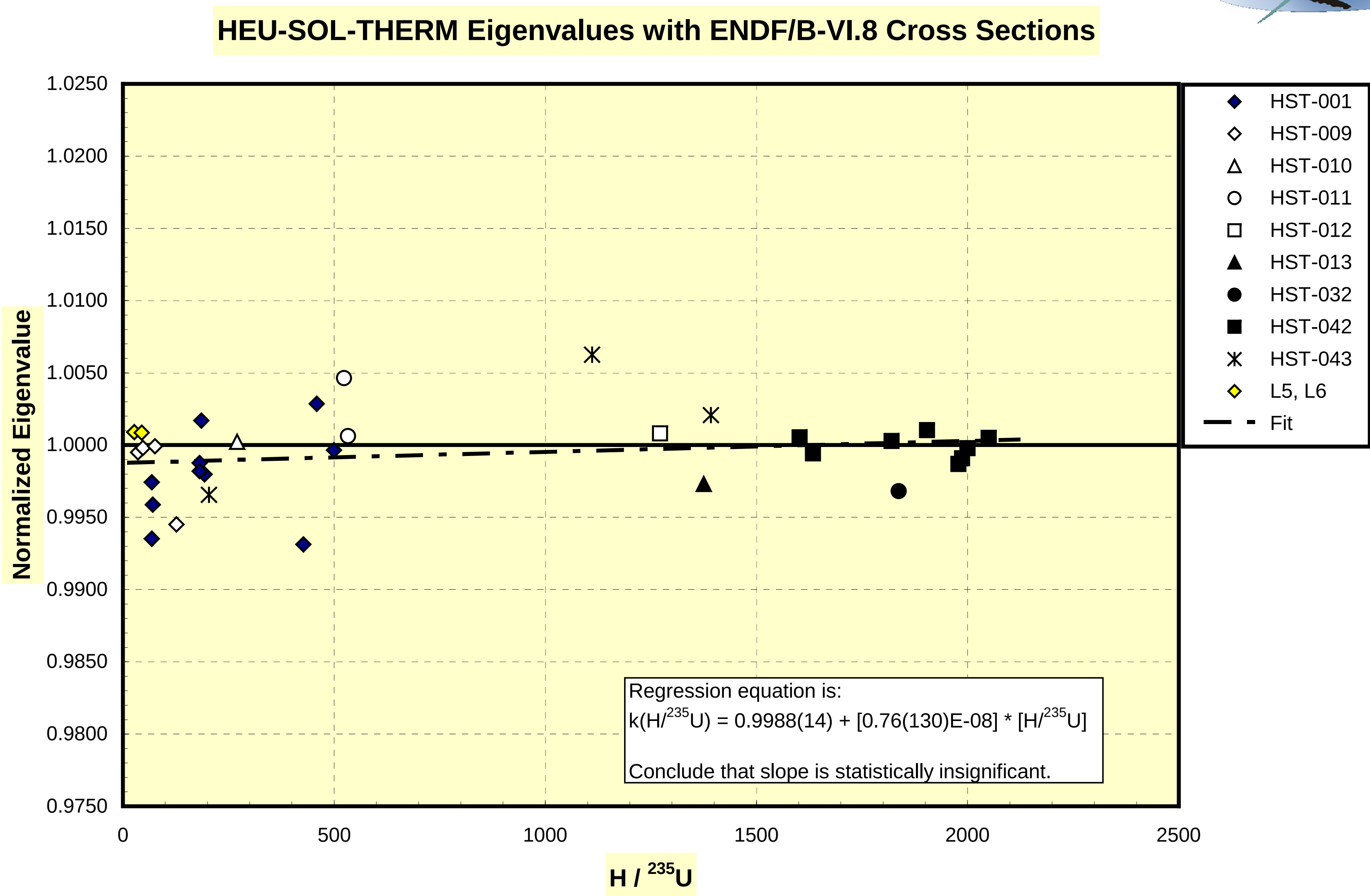
Final Observations

- Calculations with a suite of 72 HEU-SOL-THERM and LEU-SOL-THERM benchmarks demonstrate that accurate eigenvalues are obtained.
  - Average eigenvalues are near unity and there is no eigenvalue trend as a function of above-thermal leakage fraction.
  - This is true for the final ENDF/B-VI.8 library and remains true with the latest LANL/ORNL ^{235}U data set.
- Calculations with ICSBEP fast and intermediate spectrum benchmarks also yield near unity eigenvalues.
  - Hydrogen moderated benchmarks yield constant eigenvalues over a wide range of average fission energy (~ 30 eV to ~750 keV), although an unexplained reflector bias of ~0.3% is present.
  - An eigenvalue trend in excess of 1% is observed in the ZEUS/graphite benchmark (trend with AENCF?, with C/ ^{235}U?, affect of Cu?, …).
- ^{235}U $\nu(E)$ should be re-examined to see if ad-hoc changes near 1 eV to 2 eV and above 10 keV should remain.


HEU-SOL-THERM Eigenvalues with ENDF/B-VI.8 Cross Sections
Normalized Eigenvalue
1.0250
1.0200
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
0.9800
0.9750
0.00
0.10
0.20
0.30
0.40
0.50
Above-Thermal Fission Fraction ATFF)
HST-001
HST-009
HST-010
HST-011
HST-012
HST-013
HST-032
HST-042
HST-043
L5
L6
Fit
Regression equation is:
k(ATFF) = 0.9998(14) - 0.0045(94) * ATFF
Conclude that slope is statistically insignificant.

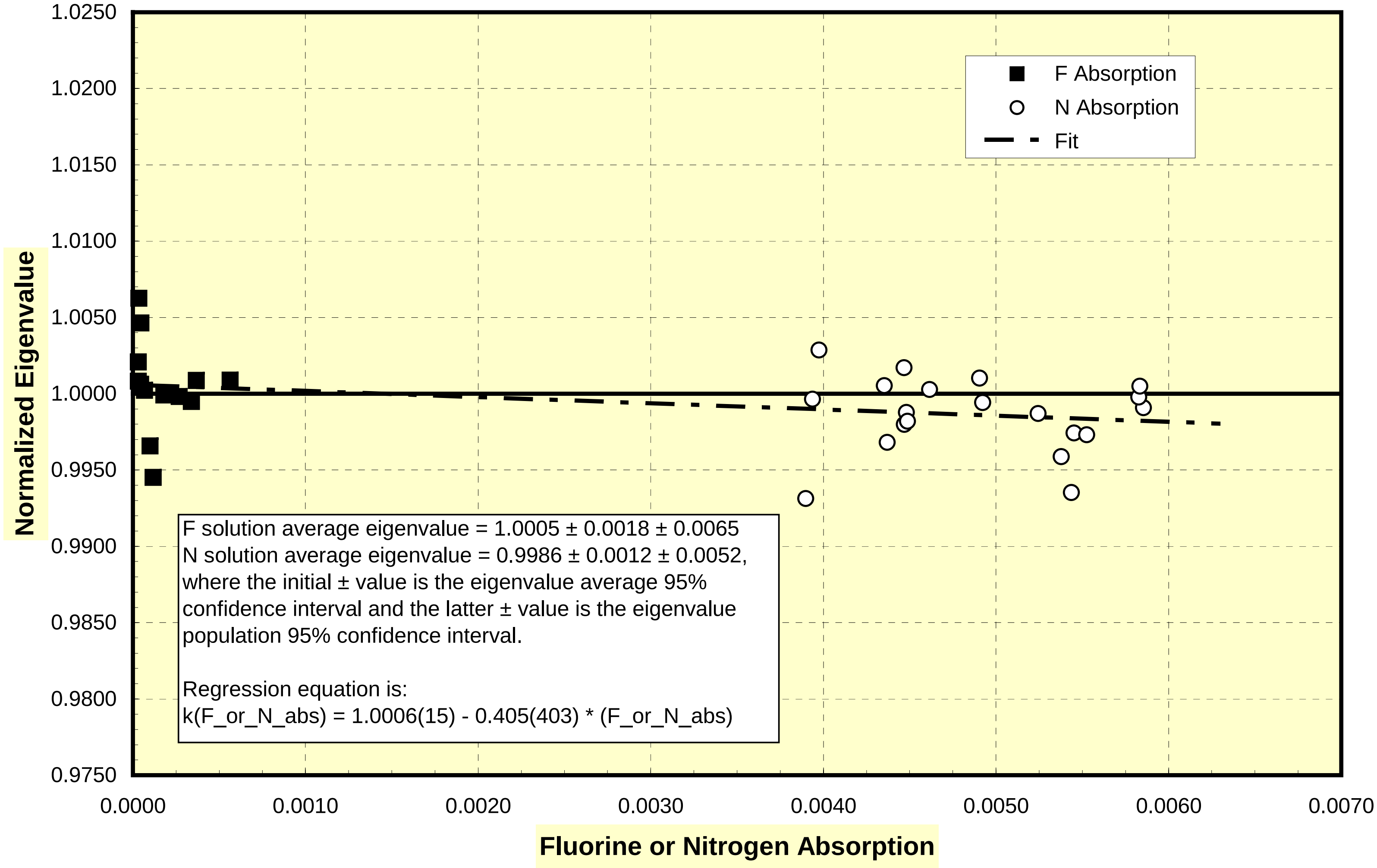
HEU-SOL-THERM Eigenvalues with ENDF/B-VI.8 Cross Sections

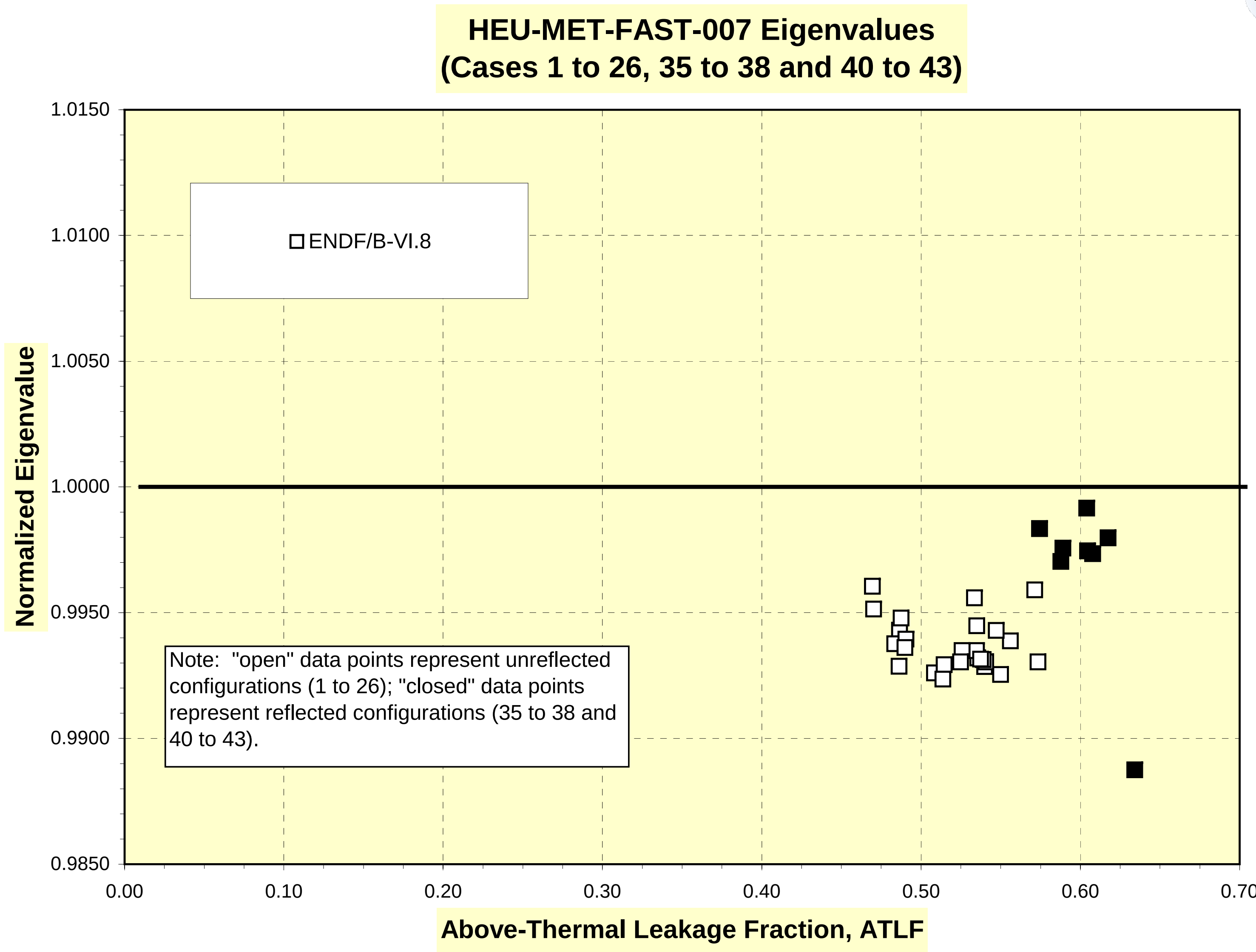




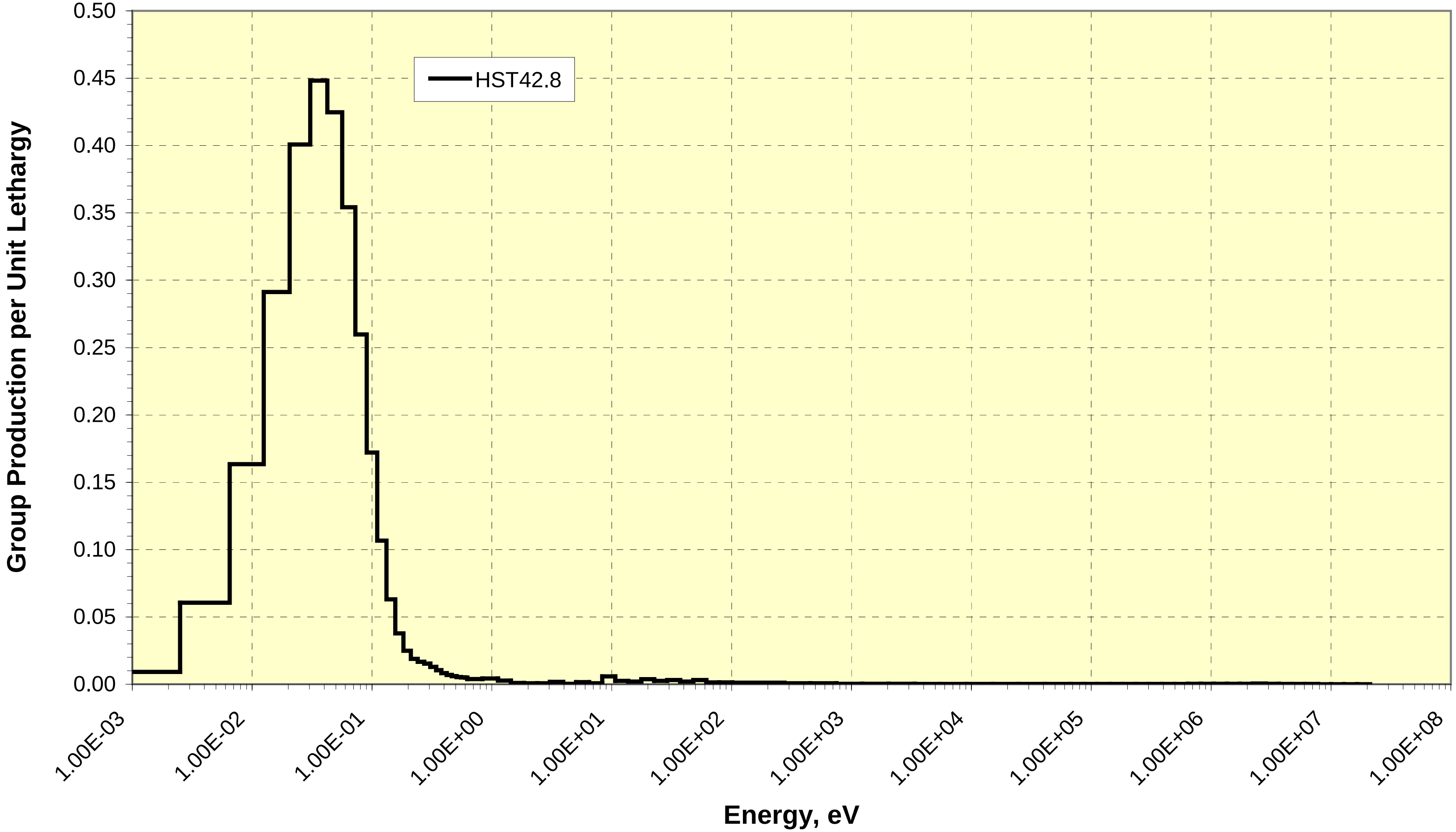
HEU-SOL-THERM Eigenvalues with ENDF/B-VI.8 Cross Sections
Normalized Eigenvalue
1.0250
1.0200
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
0.9800
0.9750
0
500
1000
1500
2000
2500
H / 235U
HST-001
HST-009
HST-010
HST-011
HST-012
HST-013
HST-032
HST-042
HST-043
L5, L6
Fit
Regression equation is:
k(H/235U) = 0.9988(14) + [0.76(130)E-08] * [H/235U]
Conclude that slope is statistically insignificant.

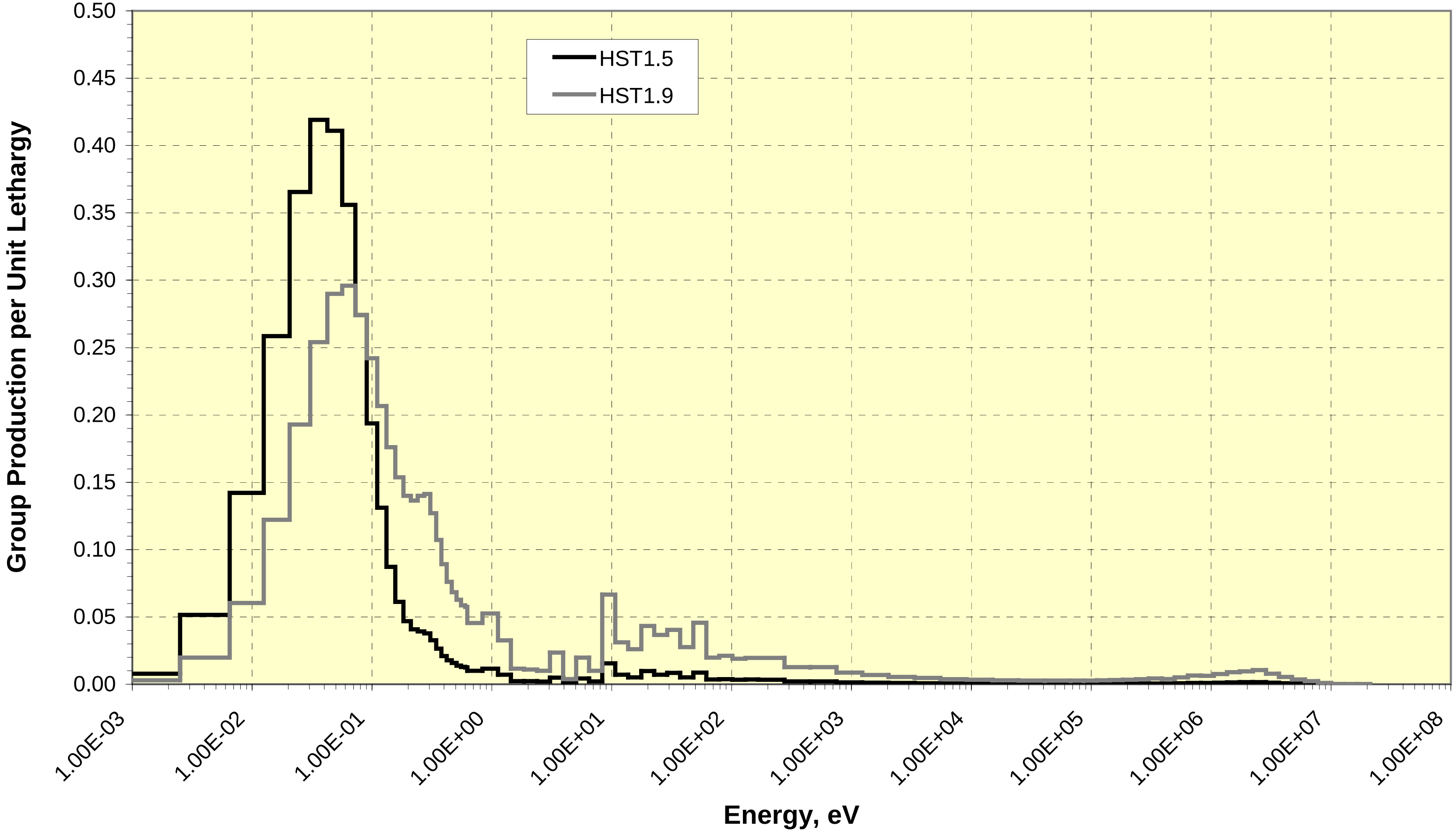
HEU-SOL-THERM Eigenvalues with ENDF/B-VI.8 Cross Sections



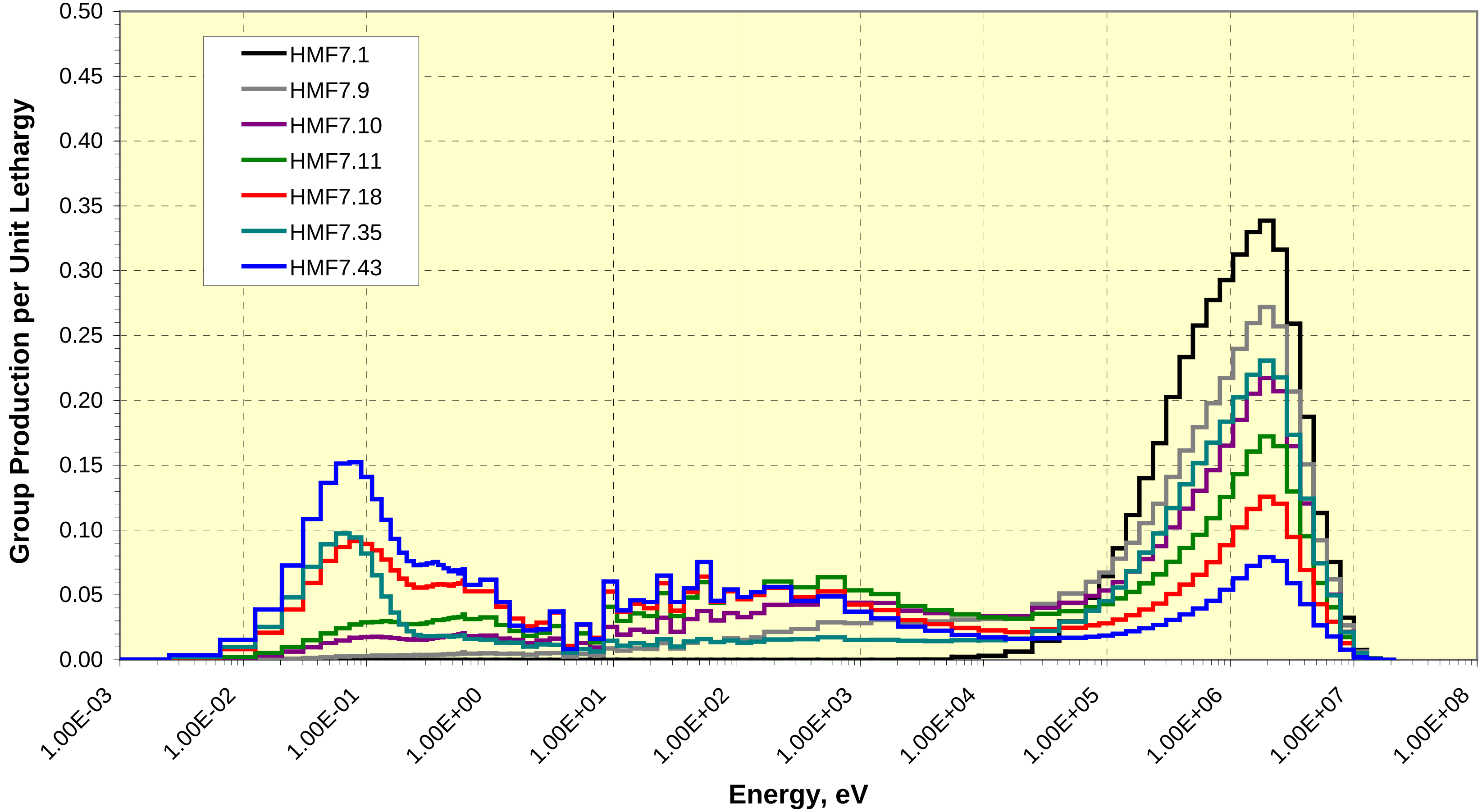


HEU-MET-FAST-007 Eigenvalues
(Cases 1 to 26, 35 to 38 and 40 to 43)
ENDF/B-VI.8
Note: "open" data points represent unreflected configurations (1 to 26); "closed" data points represent reflected configurations (35 to 38 and 40 to 43).
Normalized Eigenvalue
1.0150
1.0100
1.0050
1.0000
0.9950
0.9900
0.9850
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
Above-Thermal Leakage Fraction, ATLF

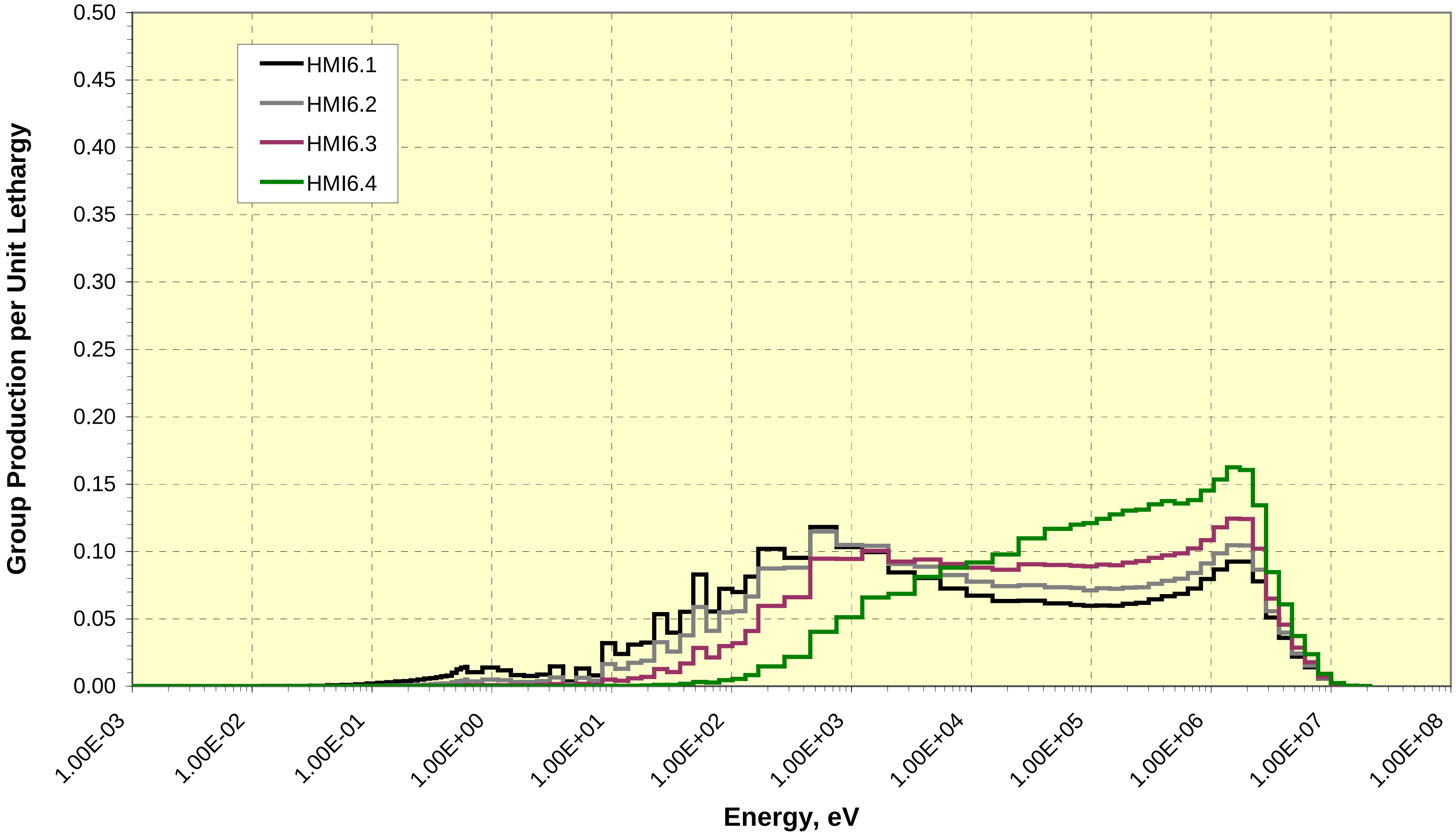

HST42.8
Group Production per Unit Lethargy
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
1.00E+07
1.00E+08
Energy, eV



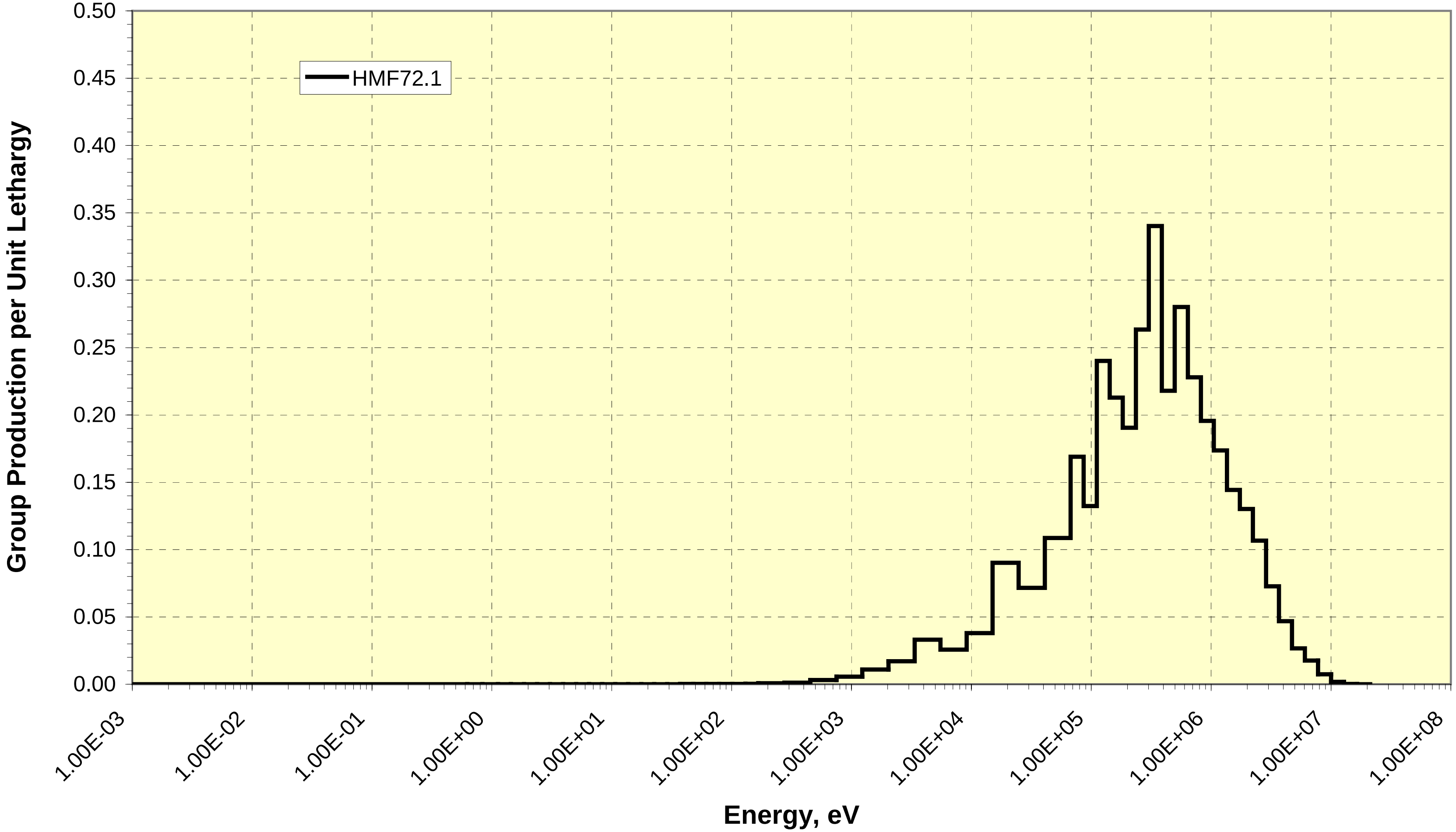
HST1.5
HST1.9
Group Production per Unit Lethargy
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
1.00E+07
1.00E+08
Energy, eV



Group Production per Unit Lethargy
Energy, eV
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
1.00E+07
1.00E+08
HMF7.1
HMF7.9
HMF7.10
HMF7.11
HMF7.18
HMF7.35
HMF7.43



HMI6.1
HMI6.2
HMI6.3
HMI6.4
Group Production per Unit Lethargy
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
1.00E+07
1.00E+08
Energy, eV


HMF72.1
Group Production per Unit Lethargy
0.50
0.45
0.40
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0.00
1.00E-03
1.00E-02
1.00E-01
1.00E+00
1.00E+01
1.00E+02
1.00E+03
1.00E+04
1.00E+05
1.00E+06
1.00E+07
1.00E+08
Energy, eV

BETTIS
LABORATORY