Integral Quantities Calculated from ENDF Libraries

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Evaluated Reaction Libraries

- *ENDF/B-VII.0 (USA, 2006)*
- *JEFF-3.1 (Europe, 2005)*
- *JENDL-3.3 (Japan, 2002)*
- *ROSFOND (Russia, 2008)*
- *ENDF/B-VI.8 (USA, 2001)*

How we can quickly validate nuclear data and select better evaluations using a few benchmarks?

ENDF Benchmarks

- Selected benchmarks (observables): $\sigma^{thermal}$, $\sigma^{14\ MeV}$, Resonance Integrals, $\sigma^{Maxw(30\ keV)}$, $\sigma^{Maxw(252Cf)}$
- The reaction of interest: (n,tot), (n,el), (n,inel), (n,2n), (n,fission), (n,γ), (n,p), (n,α)
- One may compare observables for each library however good agreement makes sense only if we deal with independent evaluations
- ENDF independent evaluations: *Atlas of Neutron Resonances*, *Standards Evaluation*, KADONIS/Bao *et al.*, ROSFOND (?)

Benchmark Calculations

- N
- "E
- O
- C

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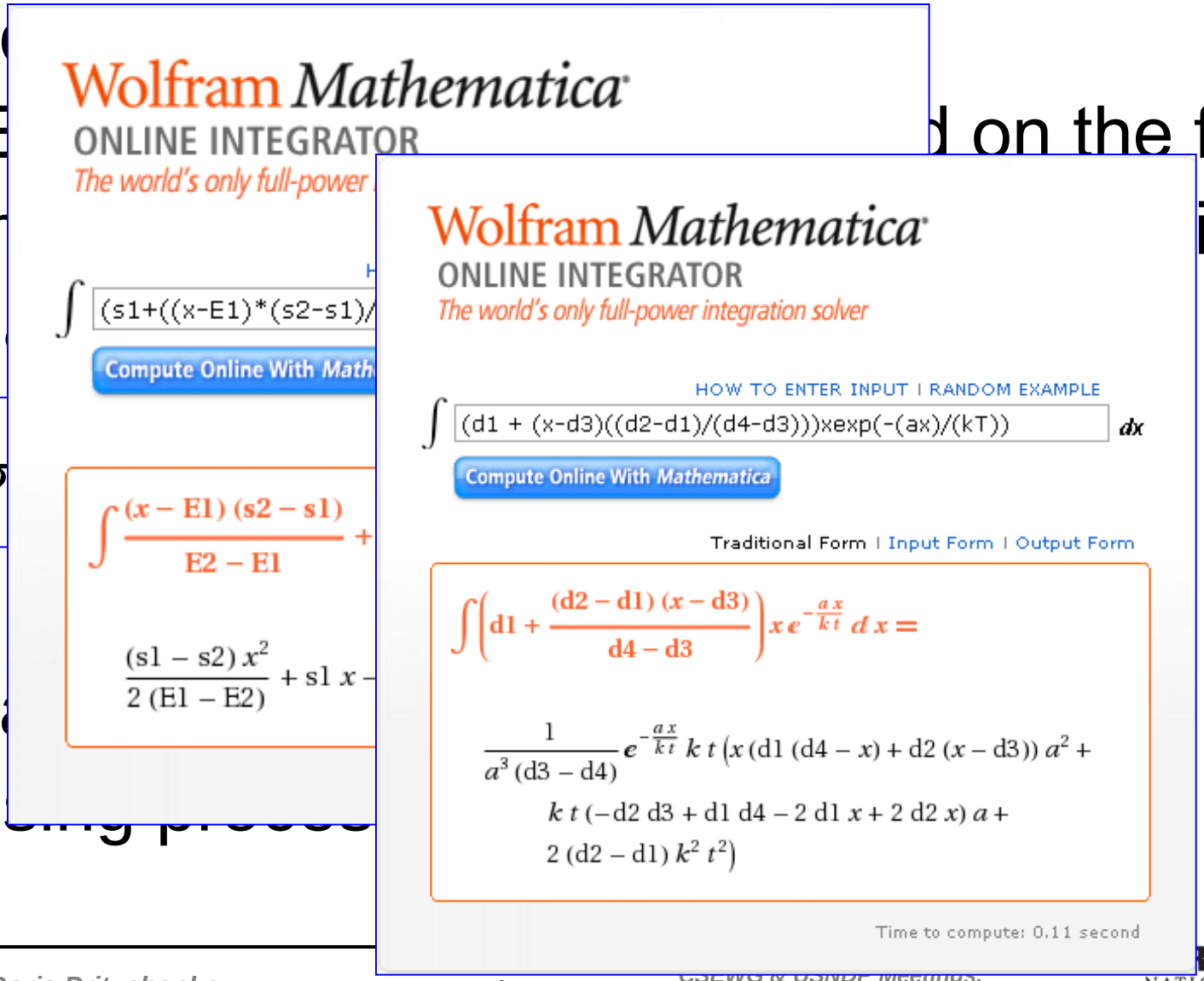
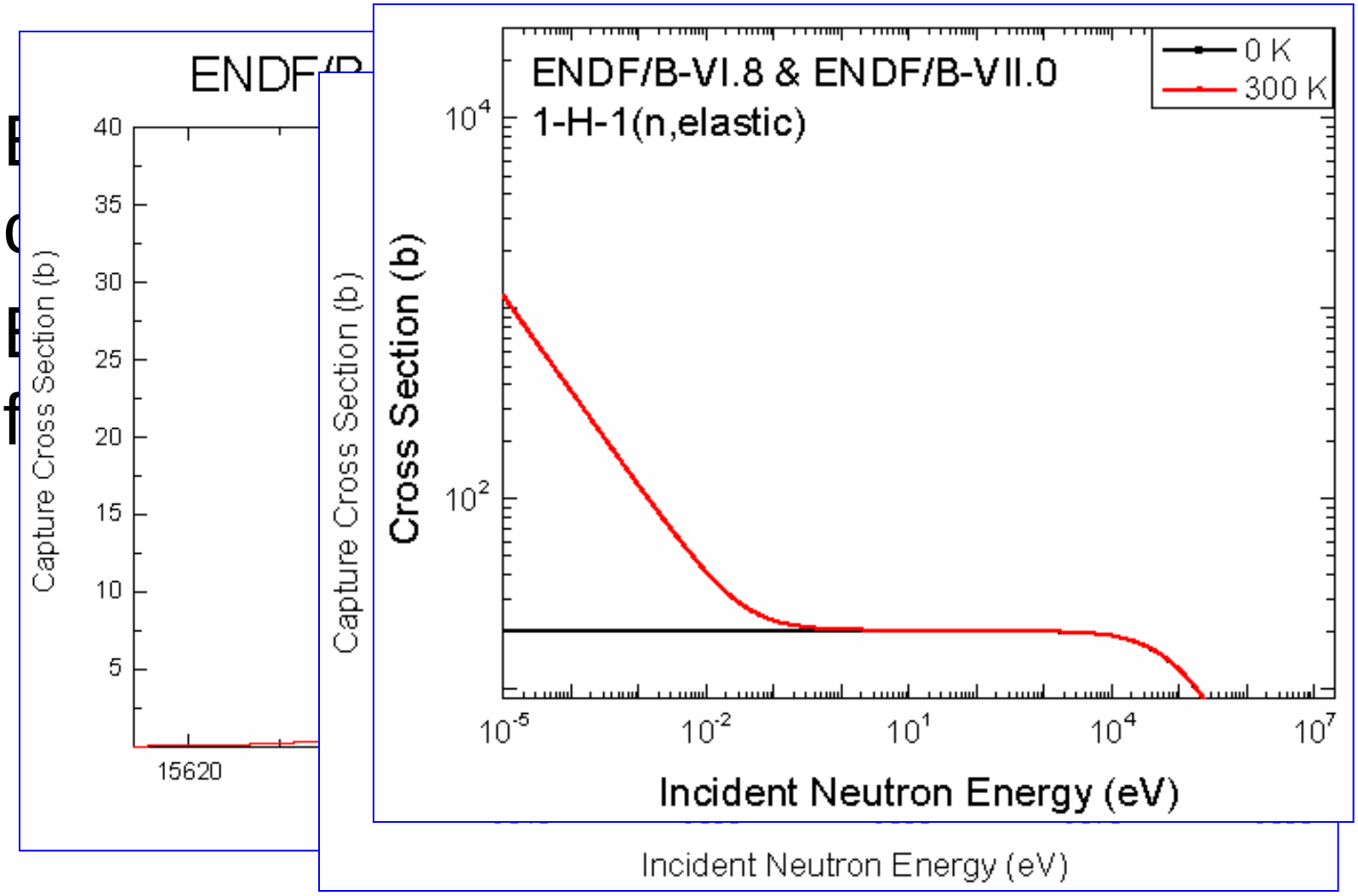
Java u



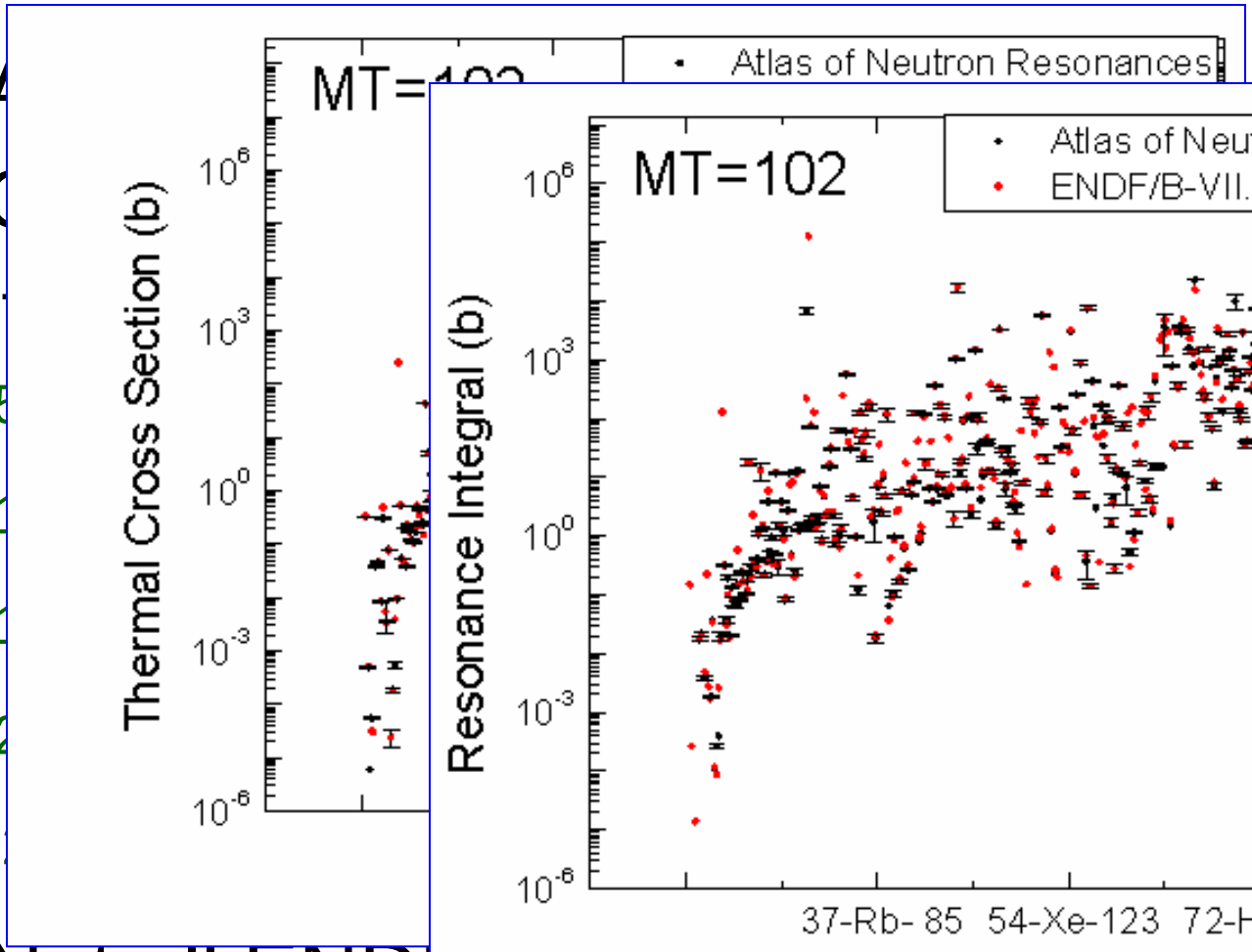
Benchmark Validation

- E
- E
  f



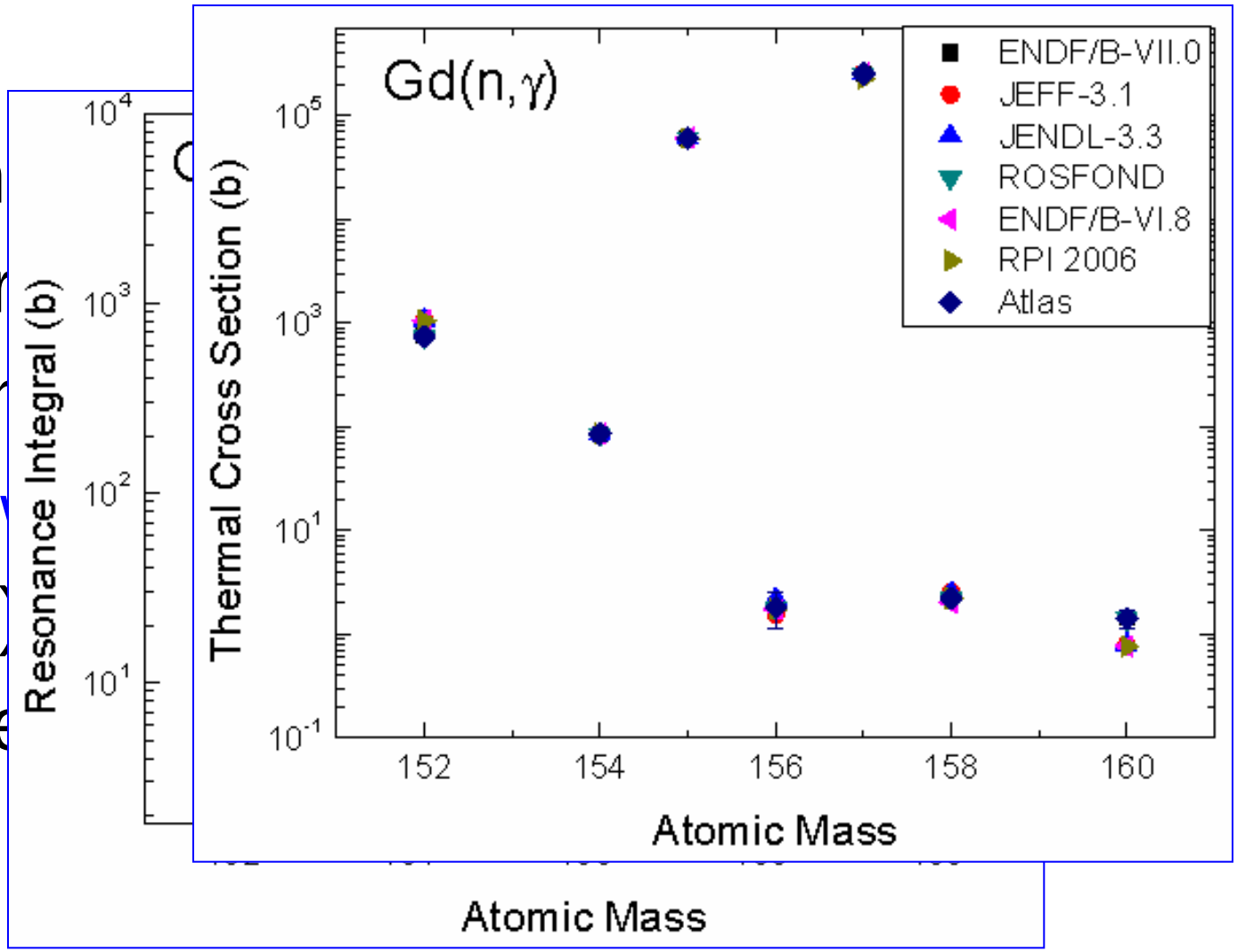
Results

- A
- C
- Not all END



Results

- Gadolin
- Compar
  of Neutr
  (G. Lein
  (2006))
- Possible

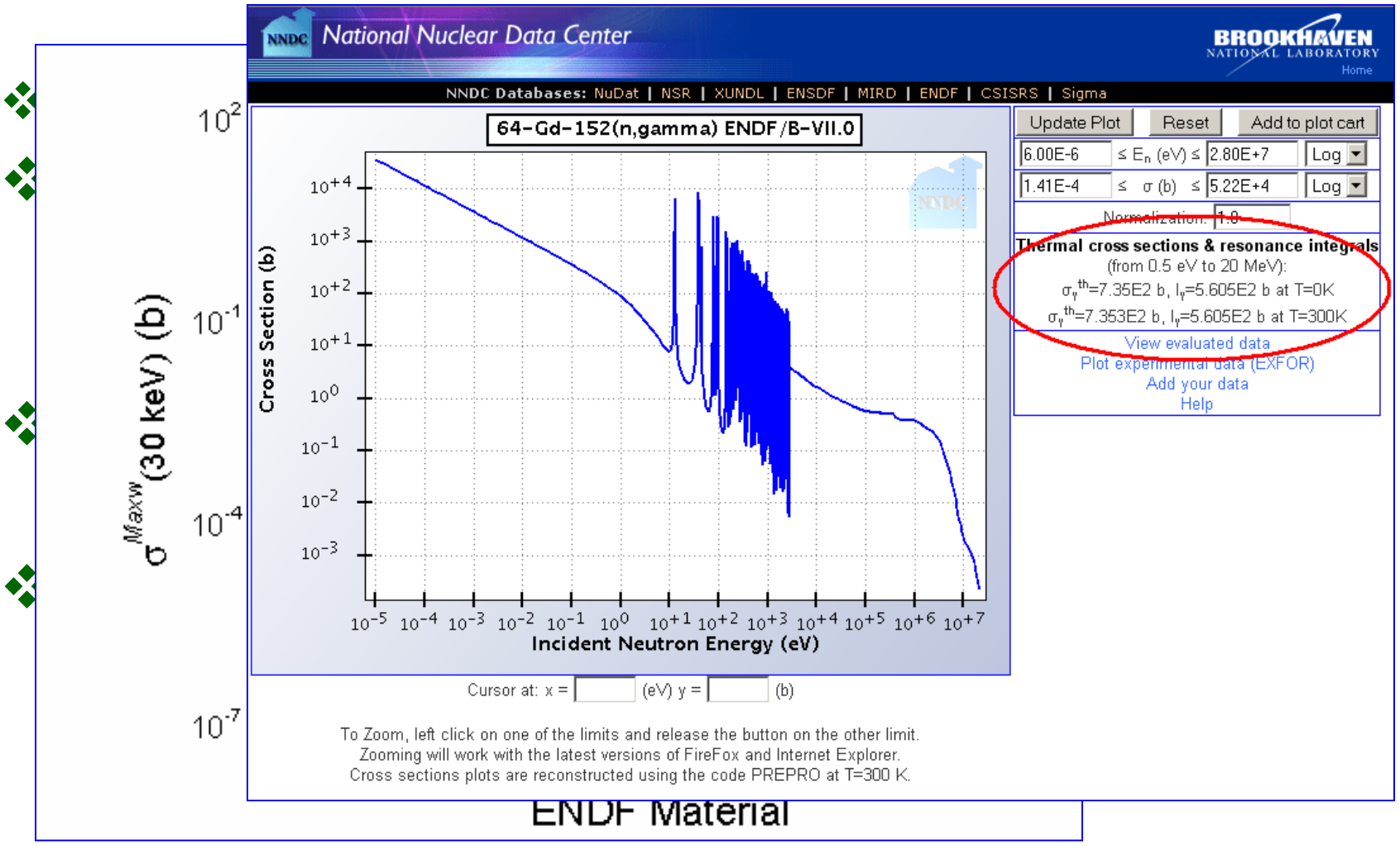


Results

Table I. ENDF/B-VII.0 pre-calculated thermal neutron cross sections and resonance integrals (0.5 eV – 20 MeV) for elastic scattering, neutron capture and fission at T = 0^0 K; corresponding values from the *standards evaluation* and *Atlas of Neutron Resonances* are shown in round and square brackets, respectively.

Material	$\sigma_{n,el}$ (b)	$I_{n,el}$ (b)	$\sigma_{n,\gamma}$ (b)	$I_{n,\gamma}$ (b)	$\sigma_{n,f}$ (b)	$I_{n,f}$ (b)
^{233}U	12.15116 (12.11(*66*)) [12.7(*3*)]	168.6988	45.23763 (45.56(*68*)) [45.5(*7*)]	141.0432 [138(*6*)]	531.2151 (531.22(*133*)) [529.1(*12*)]	775.4899 [775(*17*)]
^{235}U	15.08416 (14.087(*220*)) [14.02(*22*)]	170.07	98.68643 (99.40(*72*)) [98.8(*8*)]	140.426 [146(*6*)]	585.0856 (584.33(*102*)) [582.6(*11*)]	275.94 [275(*5*)]
^{238}U	9.279782 [9.075(*15*)]	346.8409	2.682608 (2.677(*12*)) [2.680(*19*)]	275.5847 [277(*3*)]	1.679455E-5 [3E-6]	2.691588 [0.00163(*16*)]
^{239}Pu	7.975233 (7.8(*96*)) [7.94(*36*)]	178.3488	270.3295 (271.5(*214*)) [269.3(*29*)]	181.3443 [180(*20*)]	747.4013 (750(*183*)) [748.1(*20*)]	302.5637 [303(*10*)]
^{241}Pu	11.23797 (12.13(*261*)) [9(*1*)]	175.1174	363.0489 (361.79(*496*)) [362.1(*51*)]	179.9437 [162(*8*)]	1011.852 (1013.96(*658*)) [1011.1(*62*)]	569.6337 [570(*15*)]

Results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BROOKHAVEN NATIONAL LABORATORY
NNDC Databases: NuDat | NSR | XUNDL | ENSDF | MIRD | ENDF | CSISRS | Sigma
64-Gd-152(n,gamma) ENDF/B-VII.0
Cross Section (b)
Incident Neutron Energy (eV)
Update Plot
Reset
Add to plot cart
6.00E-6 ≤ E_n (eV) ≤ 2.80E+7 Log
1.41E-4 ≤ σ (b) ≤ 5.22E+4 Log
Normalization: 1.0
Thermal cross sections & resonance integrals
(from 0.5 eV to 20 MeV):
σ_γ^th=7.35E2 b, I_γ=5.605E2 b at T=0K
σ_γ^th=7.353E2 b, I_γ=5.605E2 b at T=300K
View evaluated data
Plot experimental data (EXFOR)
Add your data
Help
Cursor at: x = (eV) y = (b)
To Zoom, left click on one of the limits and release the button on the other limit.
Zooming will work with the latest versions of FireFox and Internet Explorer.
Cross sections plots are reconstructed using the code PREPRO at T=300 K.
σ^Maxw(30 keV) (b)
ENDF Material

Conclusion

- Integral quantities have been calculated for five major ENDF Libraries and eight MT numbers
- *Comparison between calculated quantities and independent benchmarks allow to identify potential problems*
- These results are analyzed and loaded into Sigma databases
- Future plans will include extension of the current project for Westcott factor calculations
- Many thanks to M. Herman, S. Mughabghab, V. Zerkin