https://doi.org/10.61092/iaea.ac7x-p68k

INDC(NDS)-0893
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INDC International Nuclear Data Committee

The International Nuclear Data Evaluation Network (INDEN) meeting on Actinide Evaluation in the Resonance Region (5)

Summary Report of the IAEA Technical Meeting

IAEA Headquarters, Vienna Austria
20-23 November 2023

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November 2024

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ABSTRACT

An IAEA Technical Meeting on Actinide Evaluation in the Resonance Region of the International Nuclear Data Evaluation Network (INDEN) was held as a hybrid meeting from 20 to 23 November 2023. The meeting was a follow-up of the working group on evaluations in the resonance region of actinide nuclei. On-going evaluation work on U-233, U-235, Pu-239, and U-238 was discussed. Special attention was devoted to the burn-up issue and the impact of U-238 and Pu-239 RRR evaluations on burn-up calculations, with focus on the upcoming release of the ENDF/B-VIII.1 library. Particular attention was also paid to potential reference integrals for total and capture cross sections proposed for TOF fission data of fissile targets in the RRR.

November 2024

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1. INTRODUCTION

The sixth Meeting on Actinide Evaluation in the Resonance Region of the International Nuclear Data Evaluation Network (INDEN (https://www-nds.iaea.org/INDEN/)) was held from 20 to 23 November 2023. The work includes the coordination of evaluations for the upcoming nuclear data libraries (ENDF/B-VIII.1 and JEFF-4.0) and discussion of possible solutions, including those adopted by JENDL-5 evaluators. The goal is to consolidate high quality resonance data sets which also solves the integral benchmarks issues, such as inconsistent integral trends observed as a function of burnup. Yaron Danon was designated Chairman of the meeting. Dimitri Rochman and Marco Pigni agreed to act as rapporteurs. The present report summarizes the on-going evaluation work on major actinides and ^{233}U. The adopted agenda, participants list and links to participants' presentations are provided in Annex I-III, respectively.

2. PRESENTATION SUMMARIES AND DISCUSSIONS

The agenda includes various presentations on topics such as burnup calculations, actinide evaluations, validation measurements, and neutron-induced reactions.

2.1. Adjustment of ^{239,240,241}Pu for the JEFF4T3 library, D. Rochman

This presentation details the adjustment work performed for the three isotopes ^{239}Pu, ^{240}Pu and ^{241}Pu, submitted for inclusion in JEFF-4T3. The prior files for these three isotopes were the ones proposed by G. Noguere. The adjustement method is using the GLLS approach, allowing changes for the resonance widths of these three isotopes, up to 50 eV, 6.67 eV and 48.1 eV, respectively. In addition, the nubar of ^{239}Pu was allowed to change, using 14 energy groups up to 16 eV. Both differential and integral data were used for the adjustment: thermal standard cross sections for ^{239}Pu and ^{241}Pu, alpha ration for ^{241}Pu, and 18 PST benchmarks and the kritz benchmarks. Additionally, the reactivity calculation of a VERA UO2 pincell was used, taking into account the k_inf as well as the ^{239}Pu build up. The reference calculations were performed with the JEFF-3.1.1 library. After adjustment, the performances of the three isotopes were verified by running actual simulations and comparisons for the cited cases, and validation was performed with additional UO_2 and MOX pincell calculations, and Duke PWR benchmarks. Finally, the performances of the three Pu isotopes were notably improved.

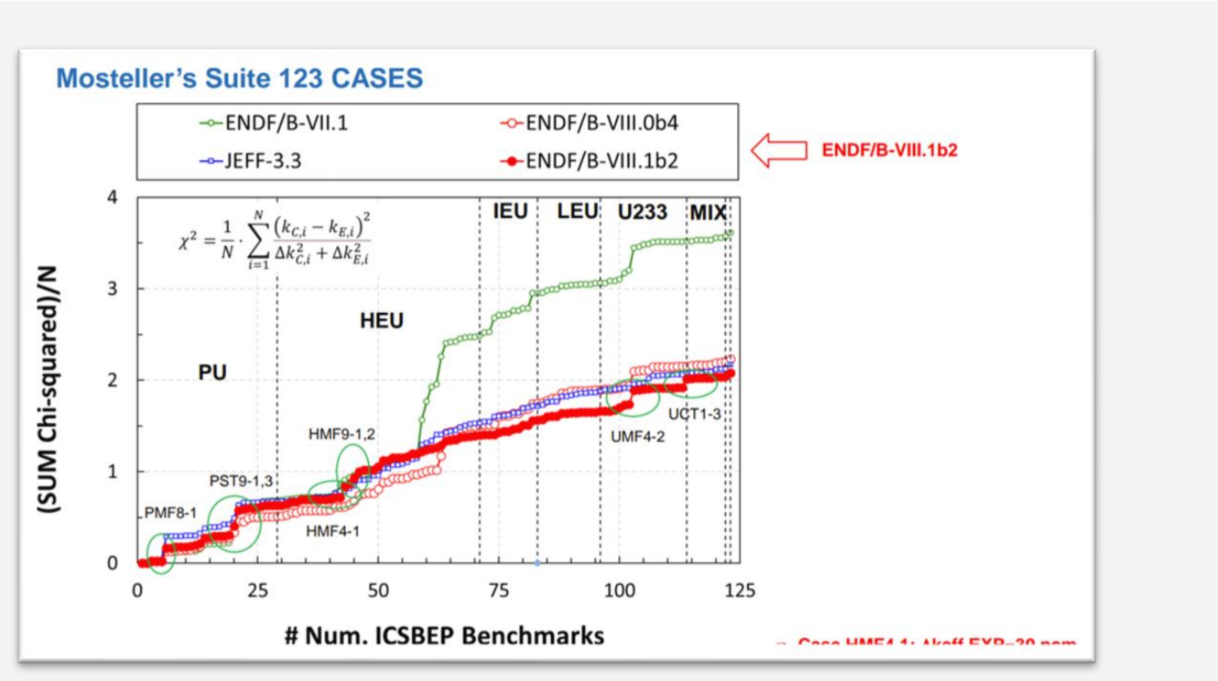
Discussion:

- There is a contradiction between well-thermalized PSTs (PST9, pST12, PST38) and the burnup curve. Increased criticality of high-leakage PST by about 500-700 pcm. This greatly worsen the achievement of WPEC subroup (VII.1) that reduced that increase almost to zero pcm from VII.0. Note that the WPEC-SG34 used the ENDF/B-VII.1 PFNS, which does not agree with current recommendations from the IAEA PFNS CRP.
- first changes to improve the loss of reactivity were performed on ^{238}U capture RRR giving +200 pcm gain at the end of burn-up.
- Two solutions. One is the above. The other one includes the change in the U-235 nubar. The way nubar is evaluated in RRR up to 100 eV is normalized to the thermal point. Here, this is not considered because the changes were done between 0-16 eV.
- Note that in the ORNL CSWEG presentation the Pu-241 nucleus was not included in the analysis.

2.2. Status of burnup calculations: ENDF/B-VIII.1b2, O. Cabellos

This presentation details the benchmarking performed for the beta2 version of the ENDF/B-VIII.1 library including criticality benchmarking,

1. Criticality Benchmarking:
   - Performance comparison of different nuclear data libraries using Mosteller's suite with 123 benchmarks.
2. Burnup Calculations:
   - Issues related to loss of reactivity at pin-cell and core levels. JEFF4T2 is not good.
   - Reactivity underestimation observed with different nuclear data libraries.
   - Significant impact due to isotopic buildup, particularly with ^{239}Pu and ^{238}U.
3. Sensitivity Analysis:
   - Analysis of sensitivity coefficients for various isotopes.
   - Impact of changes in nuclear data on reactivity predictions.
4. Reactivity Modifications:
   - Modifications in reactivity at fuel assembly and core levels.
   - Comparisons between different nuclear data libraries and their impact on reactivity.
5. Conclusions:
   - B81beta2 shows good performance in criticality benchmarking.
   - Ongoing issues with burnup calculations for several test libraries.



Discussion:

- ^{239,240,241}Pu from Gilles (unadjusted) are very close to B81beta2.
- IMF3, IMF4 are Zeus benchmarks; -600 pcm (ORNL) is too much loss.
- Although JEFF-3.3 is much worse than ENDF/B-VIII.0 in depletion calculations, ^{238}U evaluation is the same in JEFF-3.3 and ENDF-8.0, something else is playing a role.
- How do you define the difference of the loss of reactivity at each step for k-infinity?
- ^{16}O has a flat -200 pcm over the burn-up cycle. Would adopting ^{16}O from ENDF/B-VII.0 help?
- (D. Rochman) Enrichment very low (slide 23 boron let down) but results are very good.
- The two versions (Dimitri and Gilles) in JEFF-3.3 should give the same results on the depletion calculations.

2.3. Testing new nuclear data libraries, M. Hursin

The presentation summarized some work done at EPFL/PSI on the V&V of modern nuclear data library, namely a set of file candidates for release as JEFF-4T3 and the files of ENDF/B-VIII.1beta2. Verification and the associated sensitivity analysis were based on pincell depletion calculations with the deterministic code Dragon and a WIMSD formatted library generated in a consistent manner. The overestimation of the reactivity loss with the modern libraries has been reduced at least until 50 MWd/kgU. For the JEFF files, this is mainly due to large changes in the high energy domain of ^{238}U; which compensate the known deficiencies of the Pu isotopes (^{239}Pu, ^{240}Pu and ^{241}Pu). In terms of validation, the first cycle data of Fessenheim-1 NPP was modelled with Dragon and PARCS. The experimental data includes the boron letdown curve as well as the power maps at BOC. The latest ENDF/B library performs as well as JEFF-3.1.1 or ENDF/B-VII.1, at least for the first cycle. This good agreement might degrade with fuel of larger exposure. The latest JEFF library does not perform well. The experimental uncertainty of the power maps is in the 5% range; as such, this information is not useful for NDL validation; at least for the small core considered (900MWe French PWR). The effect of fuel temperature treatments (temperature profile in the pellet and resonance up scattering) on the reactivity loss were shown. It is burnup dependent and in the 100pcm range.

Discussion:

- Temperature profile, what does it mean? It is the temperature dependence across the radius of the pellets. It is important if the DBRC includes the temperature profile. To be checked.
- JENDL-5 solution >100 eV with ^{238}U is giving higher burn-up. ENDF/B-VIII.1beta solutions (up 100 eV ENDF/B-VIII.0 and >100 eV from JENDL-5)
- Why is there no symmetry in the core at zero power? This is because of the detector location. There are no noticeable differences among libraries at zero power to infer any conclusion.

2.4. Fission cross sections of ^{239}Pu measured at n_TOF, D. Cano

It is shown that the JEFF-3.3 evaluation does not agree with the n_TOF measurement. David confirmed.

Discussion:

- Is the energy calibration reliable? Likely n_TOF calibrated well. N_TOF data normalized to 17-18 eV to match the DANCE data.
- n_TOF: there is a difference at thermal with other libraries? Problem is being studied.
- (R. Capote) alpha is measured. Capture cross section is proportional to the fission cross section uncertainty.
- (E. Leal) Where are the detectors placed? Inside the capture detector. The shift of the first resonance is extremely sensitive to both depletion and criticality.

2.5. Loss of reactivity in burnup calculations using recent data libraries, A. Trkov

Notes on the definition of burnup: Energy released per fission is the key quantity that defines the burnup scale. ENDF files provide the Q_f value from fission in MF=3/MT=18. They also provide split contributions to the energy released by fission fragments, neutrons, fission gammas, neutrinos, etc. in MF=1/MT=458. In addition, contributions from other reactions and from the decay of fission and capture products should be included.

Different assumptions are employed in different codes that also include the energy from decaying fission and capture products. One of the options in Serpent recommends the effective energy released per fission Ef=202.27 MeV for ^{235}U, which approximately accounts for such effects. Serpent defines the scaling factor f=Ef/Qf and scales the Q values read from MF=3/MT=18 for other fissile nuclides by this factor. OpenMC has the option to follow the same procedure.

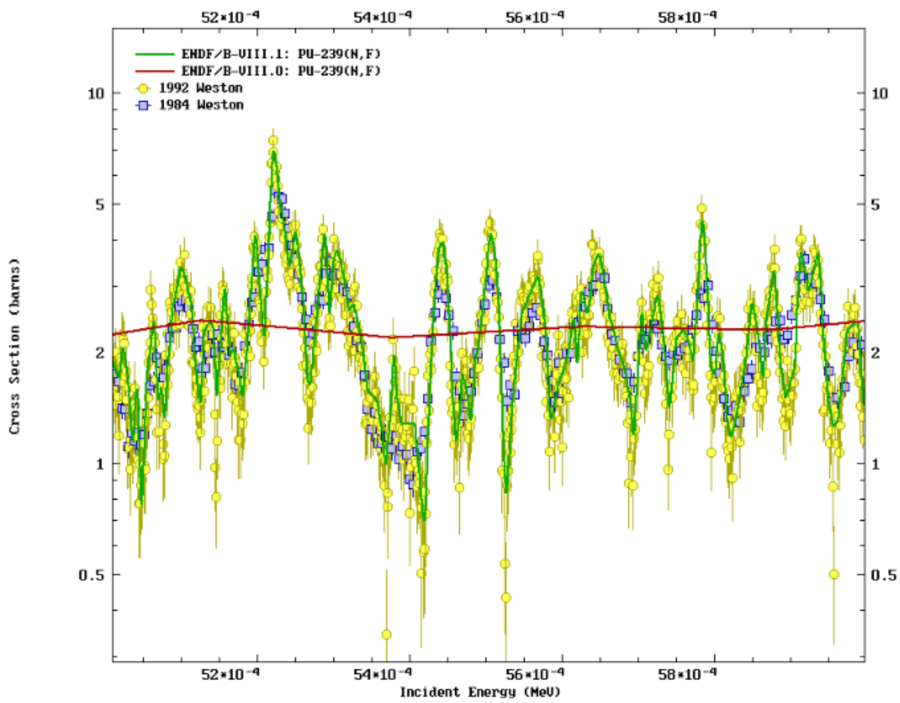
In the JSI calculations for the burnup benchmark the same list of energies per fission was used for all libraries to separate out the effects of the burnup definition and the cross sections. It seems this is the main reason for the differences in the JSI results compared to the results by other participants.

Discussion:

- What is the advantage to normalize at 500 M/T? All fission product yields are in equilibrium. To remove the effect of initial reactivity.
- Differences between Andrej's methodology to generate burn-up calculations and other ones like Oscar's. Problems may be related to the processing of VII.1 included in OpenMC.

2.6. Actinide evaluations below 100 keV: toward ENDF/B-VIII.1, R. Capote

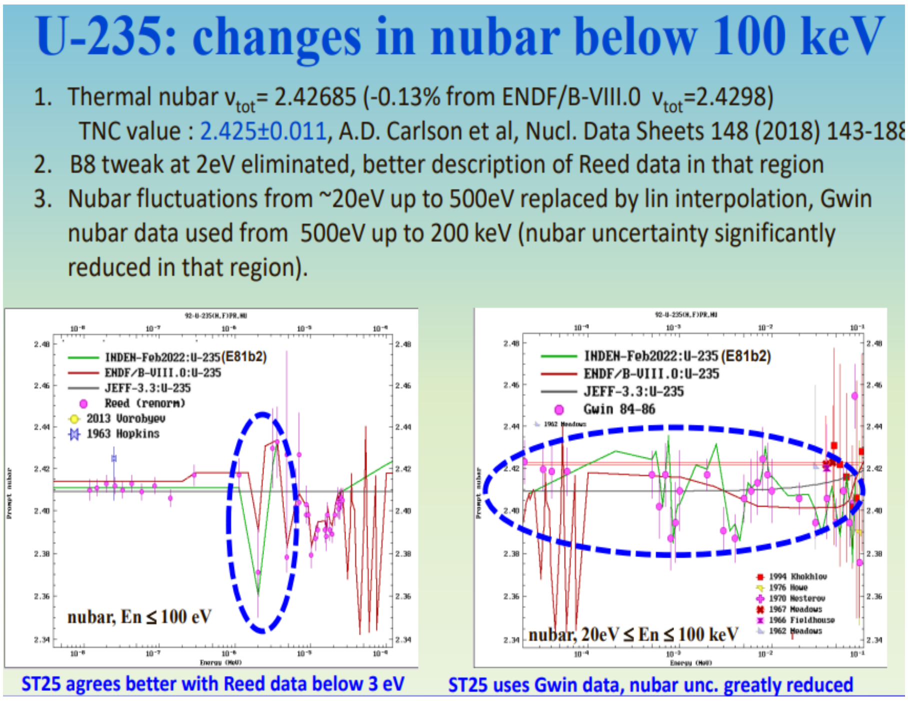
A review of INDEN actinide evaluations contributed to the ENDF/B-VIII.1 library is presented. Adoption of Weston fluctuating cross sections in the ^{235}U and ^{239}Pu URR is highlighted as shown below for ^{239}Pu(n,f).



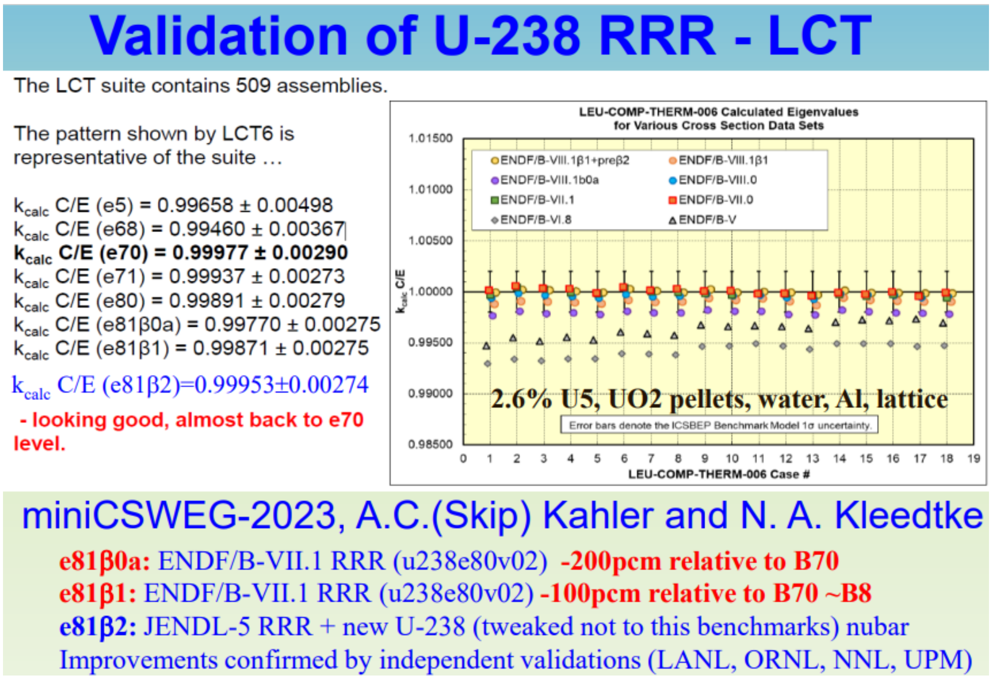
Review of newly employed PFNS at the thermal point for ^{233}U and ^{239}Pu is given. The PFNS average energy at the thermal point was reduced to 2.03 (from 2.074 MeV for the ENDF/B-VIII.0) for the ^{233}U PFNS, and to 2.074 MeV (from 2.112 MeV for the ENDF/B-VIII.0) for the ^{239}Pu thermal neutron induced fission. Additional integral constraints for the normalization of fission data measured by ToF will be published by Duran et al in Nucl. Data Sheets. A review of the compliance of current evaluations with Thermal Neutron Constants (Carlson et al 2018) was also provided.

New normalization and the mentioned PFNS changes require changes in the resonance parameter evaluations, which were discussed in the presentation (see also Pigni's talk). Updated ^{235}U resonance evaluation is shown to agree better with RPI capture data from 0.06 eV up to 11 eV measured by Danon et al shown in the INDC(NDS)-810 report of our previous meeting.

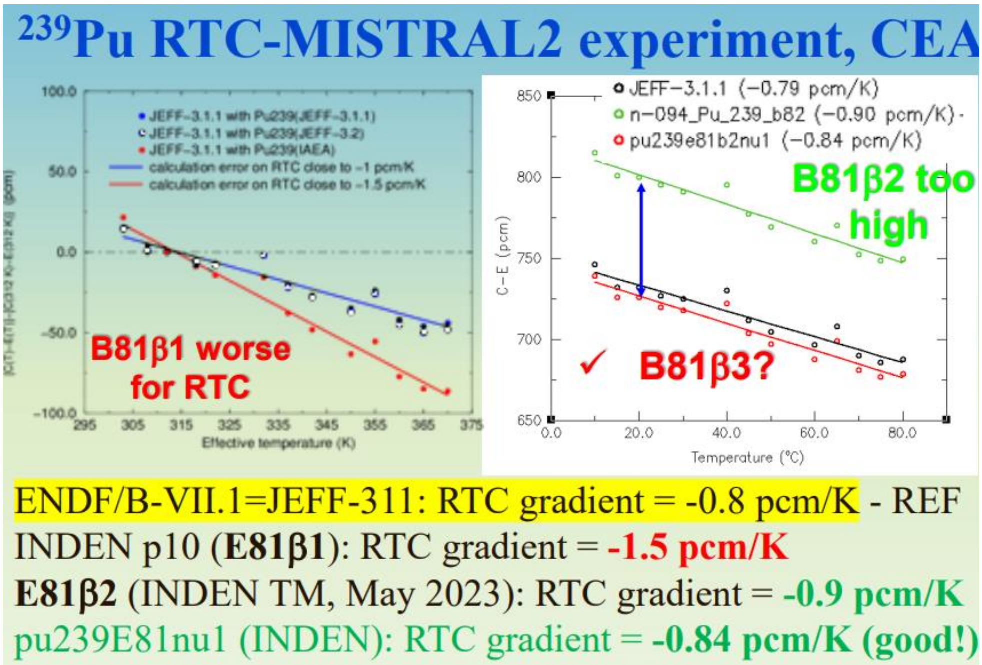
Additional changes in the neutron multiplicities of ^{233}U, ^{239}Pu, and ^{235}U nubar between 500 eV and ~80 keV were proposed based on Gwin unique measurements. An example of the ^{235}U new INDEN evaluation is shown in the figure below.



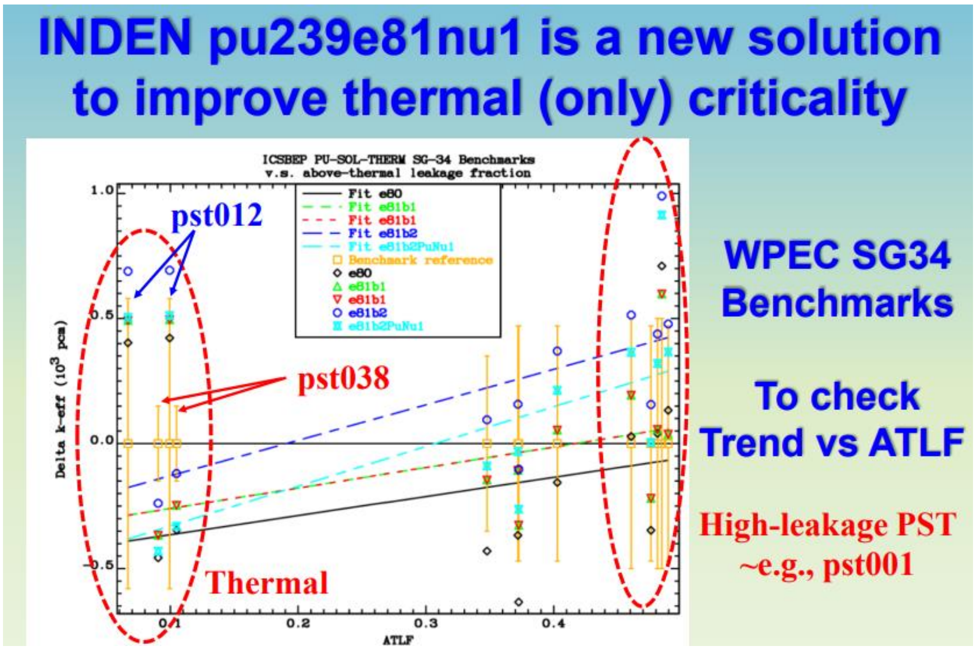
Changes undertaken in the ^{238}U neutron multiplicity were also reviewed. New ^{238}U multiplicity and updated ^{235}U resonance parameters led to an improvement of the LCT benchmarks as shown in the next slide (calculated by S. Kahler and N. Kleedtke).



Finally, the current status of the depletion issue is shown. Emphasis was made on improving the alpha description of ^{239}Pu cross sections from the thermal point up to the energy of the first resonance. ENDF/B-VII.1 is shown to exhibit the lowest subthermal alpha, which need to be matched. This results in an improvement of the reactivity temperature coefficient (RTC), which was deficient for ^{239}Pu in B81beta1 (INDEN first iteration). The beta2 criticality was still too high and was reduced to finally produce the nubar used in the ENDF/B-VIII.1beta3 which was adopted for the library release.



Criticality of PST benchmarks employed in WPEC SG34 testing still shows a gradient (e81b2Punu1) and does not reproduce the PST038, but it is the one accepted for the INDEN 2024 evaluation, as shown in the figure below.



2.7. Measurement of the neutron-induced capture-to-fission cross section ratio in ^{233}U at LANSCE, E. Leal Cidoncha

The neutron-induced capture-to-fission cross section ratio of ^{233}U has been measured at the Los Alamos Neutron Science Center at Los Alamos National Laboratory in the energy range from 0.7 eV to 250 keV. The detector setup combines the Detector for Advanced Neutron Capture Experiments (DANCE) to measure γ-rays generated from both capture and fission reactions, and the neutron detector array at DANCE to measure fission neutrons. This is the first measurement of the capture-to-fission ratio between 2 and 30 keV. The evaluations are in good agreement with the results in the resolved resonance region. In both the unresolved resonance region and the fast neutron region, a lower capture-to-fission ratio is obtained in this work from 10 to 150 keV compared to current evaluations, while good agreement with the experimental data and the evaluations is found above150 keV. Statistical model calculations were performed to compare with the experimental data. Significantly reduced 〈γ〉 was required to reproduce the measured data.

Discussion:

- (Capote) what is the value of the integral for the fission between the energy normalization region 8.1–14.7? 682.2 b.eV (fission ENDF/B-VIII.0), 106.8 b.eV (capture ENDF/B-VIII.0). Nacho's value is 689.0 b.eV.
- Large differences in alpha. Strange energy data range above 20 keV where correction for Al filter were done.
- (Capote) Can you measure H capture? No, but I can measure tritium.

2.8. Validation measurements of neutron capture gamma cascades, Y. Danon

Danon discussed the topic of "Validation Measurements of Neutron Capture Gamma Cascades". This is part of a project at Rensselaer Polytechnic Institute (RPI) with the goal of producing experimental neutron capture gamma cascade data that can be used to validate evaluated data in ENDF files and future formats that will include more information on capture cascades in addition to the spectrum.

The measurements were performed by the RPI multiplicity detector [1] which is a 16-segment NaI large cylindrical detector with the neutron beam passing in an opening in the centerline, and a sample positioned in the center. The sample is surrounded by a B-10 ceramic liner to prevent capture of scattered neutrons in the NaI. This detector was used for numerous capture yield measurements, see for example [1, 2, 3].

The experimental data is recorded event by event to collect the energy deposited in each of the 16 detector segments for every neutron capture (or fission) in the sample and as a function of neutron time of flight (TOF). The data can be processed for different incident neutron energies from 0.01 to 3000 eV. For this report data was processed to obtain the gamma spectrum from thermal capture or at a specific resonance by applying a TOF window. The spectrum was processed for each detector and for coincidence of all detectors. Examples were provided for ^{56}Fe, ^{238}U and ^{236}U.

To compare with evaluated data, the spectrum was simulated by a modified version of MCNP 6.2 [4]. The modification included the ability to read a capture gamma cascade from a file for each simulated capture event, transport the gammas in the detector, and create an output file with the energy deposited in each detector segment. The simulation was performed in analog mode through a detailed geometry of the detector. Cascades were generated using the DICEBOX [5] with input from ENSDF [6], RIPL-3 [7], and user provided parameters. The cascade file was an input to the simulation.

Fe-56 has a nearly complete set of cascade data and was used as a test to demonstrate the capabilities of the method to both measure and simulate single and coincidence capture gamma spectrum. The results for thermal neutrons on ^{56}Fe are shown in Fig. 1, the neutron binding energy is clearly visible in the coincidence spectrum. Results for ^{235}U and ^{238}U were presented and show the possibility of using this method for actinides to sperate capture gammas from fission gammas.

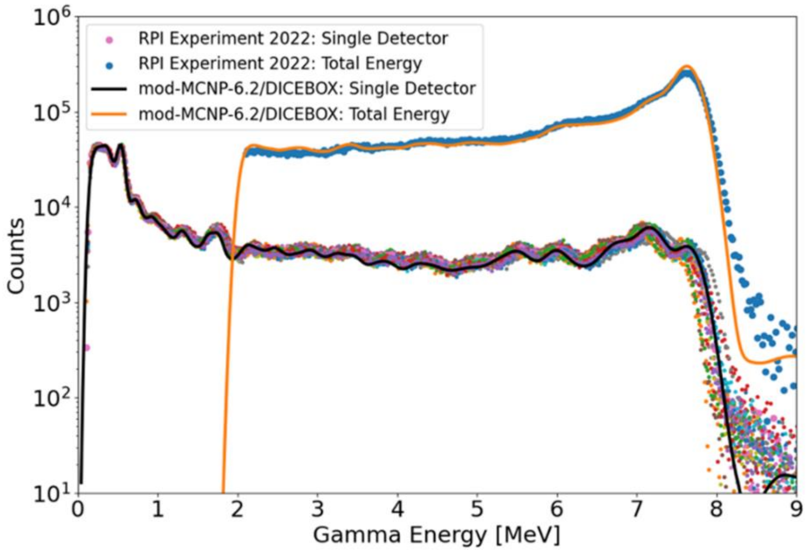


FIG. 1. Capture gamma spectrum for thermal neutron on ^{56}Fe.

The top curve is a coincidence of all 16 detectors representing the total energy deposited in the detector system. The lower curves show the spectrum to be similar in all 16 segments. The agreement between the experiment and simulation is very good except for energies above 8.2 MeV that require better handling of the energy resolution in the simulation.

References:

[1] Y. Danon, D. Williams, R. Bahran, et al., Simultaneous Measurement of ^{235}U Fission and Capture Cross Sections From 0.01 eV to 3 keV Using a Gamma Multiplicity Detector, Nucl. Sci. Eng. **187**/3, (2017) 291-301.

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[4] C. Werner, J. Bull, C. Solomon, et al., MCNP version 6.2 release notes, Technical Report LA-UR-18-20808, Los Alamos National Laboratory, Los Alamos NM, 2018.

[5] F. Becvar, Nucl. Instr. Methods A **417** (1998) 434.

[6] ENSDF, http://www.nndc.bnl.gov/ensarchivals/.

[7] R. Capote, M. Herman, P. Oblozinsky, et al., Nuclear Data Sheets **110/**12 (2009) 3107-3214.

Discussion:

- Dicebox uses ENDSF. Not all lines in ENSDF were adopted from EGAF.
- Are the gamma multiplicities known? No.
- Are the gammas coming from inelastic included? Not included. Gammas from capture for heavy elements, gamma from inelastic is usually for light nuclei. At high energy inelastic is bigger than capture.
- Elastic should be taken from the thermal. From the measured data, you can have a good estimate of elastic+capture and related uncertainty. Elastic is very small compared to the other cross section. Having elastic from the standards and uncertainties, you can derive the capture.

2.9. Neutron-induced reactions on ^{232}Th, ^{235}U and ^{238}U with thermal and fast neutrons - final results from AMS measurements, A. Wallner

Wallner used his 2022 presentation to highlight the most important aspects, see Wallner-TM-actin-2022.pdf.

2.10. New Integral references for neutron-induced fission reactions in the RRR, I. Duran

Historically, the principal international standards are a few constants at thermal point (the TNC table), including the four main fissile actinides: ^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu.

It was found useful for normalization purposes to provide integral values in standardized energy-intervals, that would help to renormalize the EXFOR datasets, before being it used at the evaluation time.

First the (n,f) reaction was studied in the range 20 to 60 meV, around the thermal point, but there are many high-resolution experiments (Tof) that start measuring at energies above 1eV, more easily reachable than the thermal point.

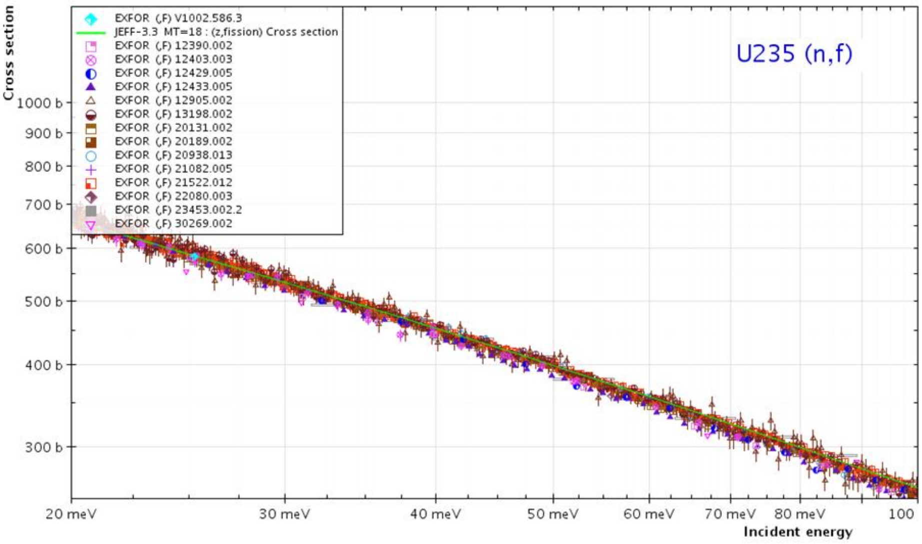


FIG. 1. Selected experimental data of $^{235}U(n,f)$ in log-log scale, after renormalization. Note that the actual slope in log-log scale is not 0.5 (that correspond to the 1/v law), being different for each actinide.

Then, new integral data on (n,f) have been proposed in the RRR, giving its ratios to the thermal point values. In the next Table are summarized: the integral values I1 obtained analytically from the fittings (20 to 60 meV) to straight lines in log-log scale with slope b and a value s at 25 meV; I3 are the integrals in the selected interval in the RRR (7.8 to 11eV, see Fig. 2); and its ratios, that are independent of the renormalizations done.

TABLE 1. INTEGRAL VALUES OBTAINED FOR FISSION ANALYTICALLY, COMPARED WITH THOSE OBTAINED FROM THE DATAFILES IN THE EVALUATED LIBRARIES.

233 U(n,f)	I_1 (b·eV)	fitted b	I_3 (b·eV)	I_3/I_1	σ(eval) (b)
This work	17.53(10)	-0.504(20)	689.0(10.8)	39.31(54)	533.0(0.7)
IAEA Standards 2017 [3]					533.0(2.2)
ENDF-B/VIII.0 [24]	17.54	-0.509	686.67	39.15	534.0
JEFF-3.3 [23]	17.45	-0.510	686.63	39.35	531.3
JENDL-4.0 [27]					531.3
235 U(n,f)					
This work	18.78(8)	-0.577(14)	245.7(4.1)	13.08(20)	586.1(2.6)
IAEA Standards 2017 [3]					587.3(1.4)
ENDF-B/VIII.0 [24]	18.93	-0.561	246.8	13.04	586.7
JEFF-3.3 [23]	18.84	-0.561	247.0	13.11	584.5
JENDL-4.0 [27]					584.5
239 Pu(n,f)					
This work	25.42(5)	-0.429(9)	1058.7(6.4)	41.65(22)	751.0(1.9)
IAEA Standards 2017 [3]					752.4(2.2)
ENDF-B/VIII.0 [24]	25.36	-0.420	1051.5	41.46	747.4
JEFF-3.3 [23]	25.33	-0.435	1073.2	42.37	749.3
JENDL-4.0 [27]					747.3
241 Pu(n,f)					
This work	34.05(47)	-0.463(31)	1377.8(33.1)	40.46(85)	1018.9(2.5)
IAEA Standards 2017 [3]					1023.6(10.8)
NDF-B/VIII.0 [24]	33.84	-0.477	1323.7	39.11	1012.3
JEFF-3.3 [23]	33.84	-0.476	1323.7	39.11	1012.3
JENDL-4.0 [27]					1012.3

Note: In between parentheses are the StdDev of the set of experiments retrieved from EXFOR.

This three-step procedure cannot be done for the (n,g) case because, as a matter of fact, the experimental datasets are scarce and/or of bad quality. Therefore the alternative way has been to obtain these integral values for the (n,tot) reaction, being the corresponding (n,g) constants deduced from the equation:

(n,g) = (n,tot) – (n,f) – (n,el)

To obtain the (n,tot) integral-values, the same procedure used for (n,f) was applied to the selected experimental data, and so we proceeded in the same three-steps way:

- Firstly, the integral values I1 and I3 for (n,tot) were obtained directly from the cross section values as found in the files retrieved from EXFOR, paying attention to the boundaries issue;
- secondly, the integral values obtained were renormalized according to the updating of the historical values used as reference, as declared by authors;
- and thirdly, the renormalized datasets in the thermal interval were used to obtain the analytical values by fitting points to straight lines in log-log scale.

Note that only those datafiles in EXFOR having point-data with high energy resolution were selected, and few outliers were discarded. Renormalization and energy calibration was eventually needed. The renormalization factor is the one obtained by updating they declared beam-flux monitor at the present days, when ^{235}U(n,f) at thermal point is 587 b (NDS18). So that, every actinide result is correlated with this U5 parameter.

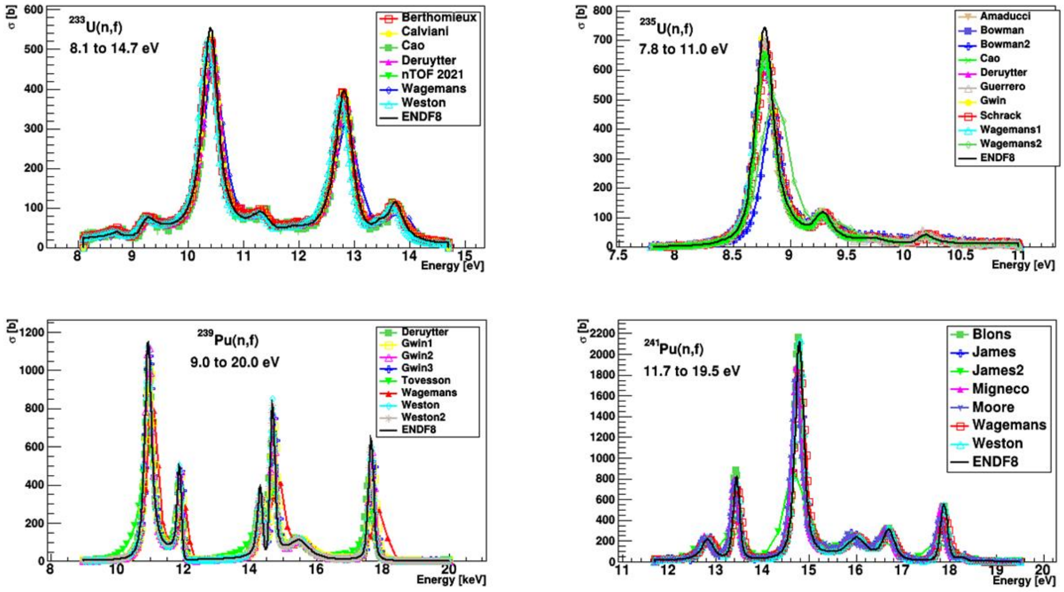


FIG. 2. Integration limits agreed to be used for the (n,f) analysis in the RRR. The same intervals have been adopted for (n,tot) and (n,g).

Once the (n,tot) parameters were obtained, both (n,el) and (n,f) parameters were subtracted to get the (n,g) ones. Concerning the (n,el) values, the experimental sources are very scarce in this energy ranges, introducing so an important uncertainty. Nevertheless, its effect is low because the (n,el) cross sections are much lower than the others. Moreover, the (n,el) evaluations are strongly correlated with (n,tot) and (n,f).The here used (n,el) XS at thermal point comes from the TNC table, but taken also into account the information given by other experimental compilations shown in the following Table:

σ(n,el) [b]	U233	U235	Pu239	Pu241
This work	12.4(0.5)	14.3(0.5)	8.0(1.0)	11.5(1.5)
TNC table	12.2(0.7)	14.1(0.2)	7.8(1.0)	11.9(2.6)
Eval.Libraries	12.2	14.1-15.1	7.8-8.8	11.3
Mughabghab	12.7	14.2	7.9	8.9
Divadeenam	12.6	14.0	7.3	9.1

Looking at the available experimental data the TNC values have been found to be slightly low but for ^{241}Pu that seems to be too high. Nevertheless, the evaluation done in this work remains well inside the uncertainties quoted in the TNC and very close to libraries.

In the next Table the thermal region parameters are summarized, compared with the values included in the TNC table:

——	σtot	I1tot	Ratiotot	σfis	I1fis	Ratiofis	σel	I1el	Ratioel	σcap	I1cap	Ratiocap
U3	590.2	19.47	30.31	533.0	17.53	30.40	12.4	0.49	25.90	44.8	1.45	30.84
TNC	590.1			533.0			12.2			44.9		
	0%			0%			+1.6%			-0.2%		
U5	700.7	22.49	31.16	586.1	18.78	31.20	14.3	0.56	25.54	100.3	3.14	31.91
TNC	700.9			587.3			14.1			99.5		
	0%			-0.2%			+1.4%			+0.8%		
Pu9	1028,7	35.21	29.21	751.0	25.41	29.56	8.0	0.31	25.81	269.7	9.48	28.44
TNC	1030.8			752.4			7.8			269.8		
	-0.2%			-0.2%			+2.6%			-0%		
Pu1	1392.1	45.87	30.35	1018.9	34.04	29.93	11.5	0.46	25.43	361.7	11.37	31.79
TNC	1397.8			1023.6			11.9			362.3		
	-0.4%			-0.5%			-3.5%			-0.2%		

Note that the TNC stot is taken by summing up (n,f) + (n,el) + (n,g). The values found in this evaluation work are very close to the TNC ones, but for ^{241}Pu where the experimental data available is relatively low. This is supporting the here found transmission values that, conversely to the capture case, are grounded on solid experimental data. In red are the percentage differences showing the general consistency of the data, even though the inverse correlation of (n,g) sigmas with the (n,el) ones, that suggest taking its sum as the value to be discussed.

In the next Table are the values of scap and I1 here obtained, and their ratios, compared with the ones found in the evaluated libraries. This ratio is of interest because it is independent of differences in the normalizations. It is worth noting the good agreement found, even when there is in ENDF8 for ^{233}U(n,g) a remarkable difference in the separate values (in red).

σ / I1(n,g)	U233	U235	Pu239	Pu241
This work	44.8(2.2)/1.45(13)=30.8	100.3(2.5)/3.14(12)=31.9	269.5(2.4)/9.5(2)=28.4	361.7(3.6)/11.4(5)=31.8
ENDF8	42.3 / 1.41 = 29.9	99.4 / 3.17 = 31.4	270.1 / 9.63 = 28.1	363.0 / 11.4 = 31.8
JEFF3.3	45.3 / 1.51 = 29.9	99.6 / 3.20 = 31.1	271.4 / 9.72 = 27.9	363.0 / 11.4 = 31.8
JENDL4	45.3 / 1.51 29.9	98.7 / 3.09 = 31.9	271.5 / 9.71 = 27.8	363.0 / 11.4 = 31.8

In order to cross-check these scap values at thermal point, that have been obtained indirectly from the (n,tot) ones, let us look at the a-values, defined as scap /sfis, comparing it with the TNC ones (NDS18) as well as from others found in the literature, as can be seen in the next Table:

Alfa-value	This work	NDS18 [ ]	Pronyaev GMA [ ]	Lounsbury [ ]	Adamchuk [ ]
U233	0.0845(41)	0.0842(17)	0.0784(32)	0.0861(21)	
U235	0.1712(43)	0.1694(22)	0.1691(35)	0.1697(29)	0.1690(35)
Pu239	0.3588(34)	0.3586(35)	0.3595(42)	0.3555(57)	
Pu241	0.3548(36)	0.3537(79)	0.3533(71)		

The values found here are systematically higher than the Standards in the TNC table, but the percentage differences are kept below 1% in part due to the differences found for the (n,f) values, that are likely slightly high in the TNC table.

Summary of (n,tot) and (n,g) integral values

The final goal of this work was to propose the integral values I1 and I3 for capture, having found a much better approach to get it by subtracting the (n,f) and (n,el) values from the (n,tot) ones obtained from selected high-resolution experimental datafiles, by the same procedure as the (n,f) ones were formerly found.

(n,tot)	I1_renorm	I1_analytical	I3_numerical	I3_renorm	I3 / I1
U233	19.5(0.1) 0.3%	19.5(0.1) 0.3%	865.6(8.3) 1%	871.2(2.5) 0.3%	44.8 0.5%
U235	22.5(0.1) 0.3%	22.5(0.1) 0.4%	372(14) 3.7%	375(10) 2.7%	16.7 3%
Pu239	35.2(0.1) 0.3%	35.2(0.2) 0.4%	1819(42) 2.3%	1834(38) 2.1%	52.1 2.5%
Pu241	45.9(0.1) 0.2%	45.9(0.1) 0.1%	2220(132) 6%	2235(94) 4.2%	48.7 4.5%
(n,g)	**I1_proposed**	**I3_evalDB**	**I3_analytical**	**I3_proposed**	**I3 / I1**
U233	1.46(13) 9%	101.3	99(12) 12%	100(9) 9%	68.5 12%
U235	3.14(12) 3.8%	93.5	92(11) 12%	93(11) 12%	29.6 13%
Pu239	9.48(17) 1.8%	680.5	664(39) 5.9%	673(39) 5.7%	71.0 6.5%
Pu241	11.5(5) 4.4%	829.0	732(100) 14%	775(98) 13%	67.4 15%

2.11. Evaluating fission of actinides from 1 MeV to 300 MeV, I. Duran

It is proposed that the evaluation of the fission cross sections on actinide targets can be normalized to measured fission cross sections in the second-fission plateau from 9 to 11 MeV. Cross sections in this plateau are higher and flatter for almost all actinide nuclei. A proper evaluation of the cross sections in this energy region can be derived from standards for ^{235}U and ^{238}U. Evaluations are proposed to be extended up to 200 MeV.

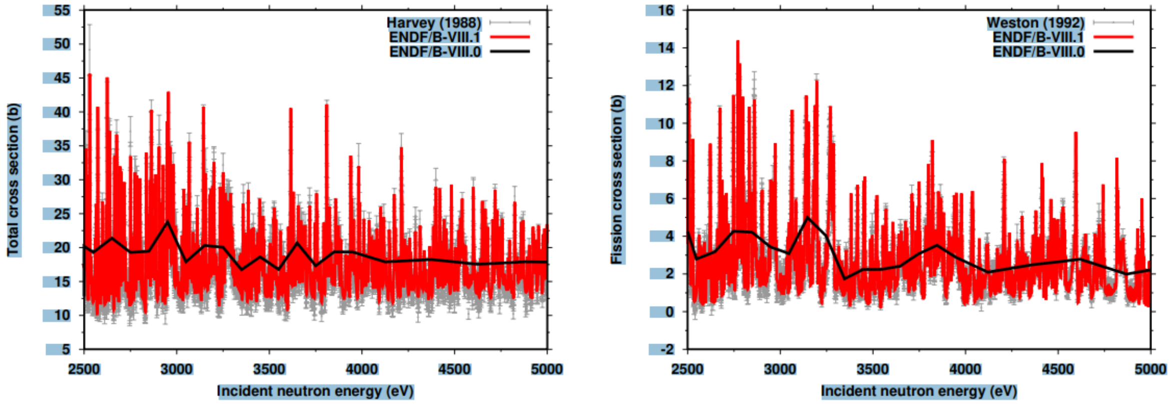
Discussion:

(Capote R.) Evaluations are driven by needs. The main reason that there are no evaluations above 20-30 MeV is because there are no clearly identified data needs above 20-30 MeV. Is it easy to evaluate? Not completely correct. If you want to extend, there is not only fission. One has to evaluate everything with scarce data. That's why many libraries are not extended. You need to have both nuclear data needs and newly measured data to update evaluations. JENDL-HE is the special case going back to 2007. At the time, the accelerator driven project in Japan was a high priority. That's why the Japanese extended evaluations and did a good job; they used some parameterization loosely based on experimental data. Standards should make the effort to extend above 200 MeV assuming we have a good evaluation of Hydrogen up to high energy. Standards are usually very different from the libraries. Neutron standards are used to do relative measurements and in evaluations. Carbon was not extended above 1.8 MeV because of a big resonance at 1.9 MeV.

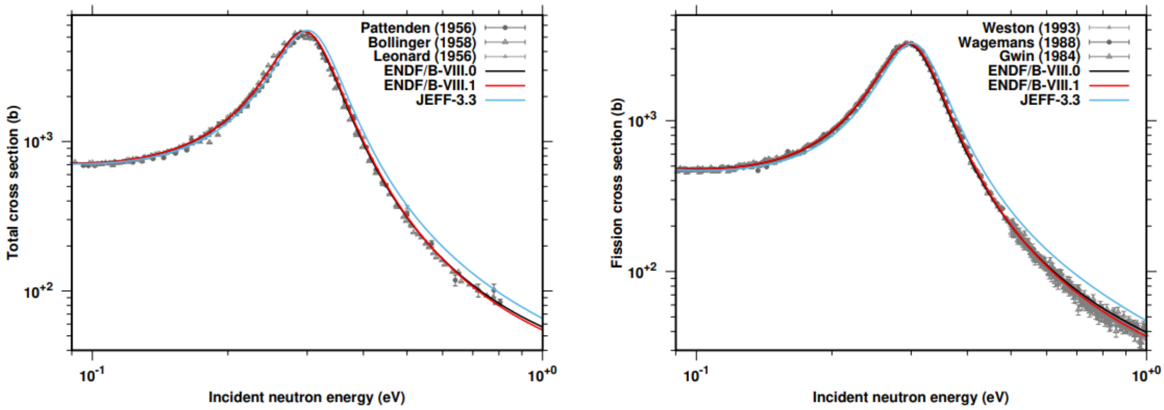
2.12. ORNL contributions to the ENDF/B-VIII.1 library, M. Pigni

The presentation highlighted extensions made to the RRR actinide evaluations toward the ENDF/B-VIII.1 release:

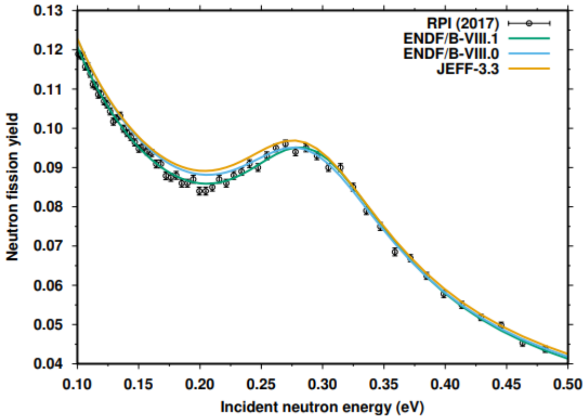
1. ^{239}Pu RRR was extended up to 5 keV using Harvey 88 (total) and Weston 92 (fission) cross section data. Note that Weston data were also used to include fluctuations in the fission cross section in the URR up to 40 keV.

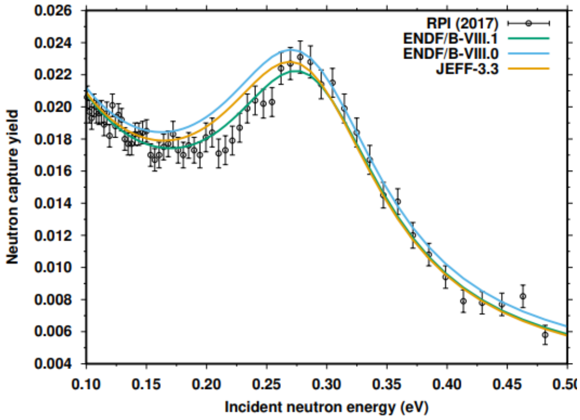


2. A potential energy shift of the 0.3eV resonance undertaken in JEFF-3.3 can improve criticality and RTC, but it is inconsistent with data as shown below.



3. ^{235}U fit of the RPI data near and below the first resonance was improved. Depletion calculations were not affected by this change.





2.13. Comment on the JENDL-5 evaluation of ^{238}U capture above 100 eV, N. Iwamoto

The adjustment method of resonance widths adopted in the evaluation of JENDL-5 is reported. The details are found in Ref. [1].

The resonance parameters in the test files (JENDL-5α1) of ^{235,238}U and ^{239}Pu were based on the CIELO-1 (ENDF/B-VIII.0). Their applications to benchmark tests of fast reactors showed lower criticalities than those of JENDL-4.0. This result suggested that the improvement of the fission, capture, and elastic scattering cross sections was needed in the fast neutron energy region. In addition, the possibility of larger capture cross section of ^{238}U in the keV energy region was indicated by time-of-flight capture measurement at CERN. According to this measurement, the grouped capture cross section of ^{238}U was increased beforehand by 2% with changing the resonance widths above 100 eV, for which a similar method described below was adopted.

To improve the performance of the integral benchmark tests for the fast reactors, the fission, capture and elastic scattering cross sections above 100 eV for ^{235,238}U and ^{239}Pu were revised. In the present evaluation, the cross sections were grouped with 100 lethargy bins in the energy region from 1 meV to 20 MeV. The adjustment of the group cross sections was performed with the generalized least square method to raise the consistency with the benchmark tests. The amount of change in the group cross sections was estimated with prior uncertainties and sensitivities to the benchmark tests. It is assumed that the prior uncertainties were energy-independent values of 1.0%, 1.5% and 2.0% for fission, capture and elastic scattering cross sections, respectively, as the group cross sections were not changed drastically. The 36 benchmark tests for the criticalities of fast reactors were adopted to revise the group cross sections, together with the five and two benchmark tests for Na-void reactivity and control-rod worth, respectively. As a result, this adjustment leads to the improvement of underestimation tendency seen in the criticality and Na-void benchmark tests with the changes of the group cross sections smaller than 1%. This improvement is mainly caused by the changes of the fission cross section of ^{235}U and capture cross section of ^{238}U.

It is difficult to adopt the adjusted group cross sections in the resolved resonance region, instead of taking resonance parameters. As the next step, the resonance widths (i.e., neutron, gamma, and fission widths) had to be changed to have the group cross sections consistent with the adjusted ones. To revise the group cross sections, factors multiplied by respective widths were introduced. They had the same values in each lethargy bin but were different for respective widths. The sensitivities of the group cross sections to the factors were prepared to facilitate the improvement of the group cross sections.

According to the adjusted results, the resonance widths were successfully revised with the sensitivities. The obtained amounts of change in neutron and gamma widths, for example, compared to the initial widths taken from ENDF/B-VIII.0 are typically smaller than 2% and 5%, respectively.

References:

[1] O. Iwamoto, N. Iwamoto, S. Kunieda, F. Minato, S. Nakayama, Y. Abe, et al., Japanese evaluated nuclear data library version 5: JENDL-5, J. Nucl. Sci. Technol., 60(1), (2023) 1-60.

Discussion:

- ^{238}U analysis done by Stefan K., Peter S., and Gilles N. Wright experiment has some problem. In ENDF/B-VIII.0 release, the RRR evaluation was based on JRC data; for different reasons, Mingrone's and Wright's data were not included in the analysis, Wright data should be revised by n_TOF. Mingrone cannot be corrected. A new measurement is planned.
- Did your adjustment produce a new covariance information? Are fission cross section and nubar correlated? No, I do not think so.
- DBRC discussion. Kang Seog and Mathiew results on DBRC are consistent.

General discussion:

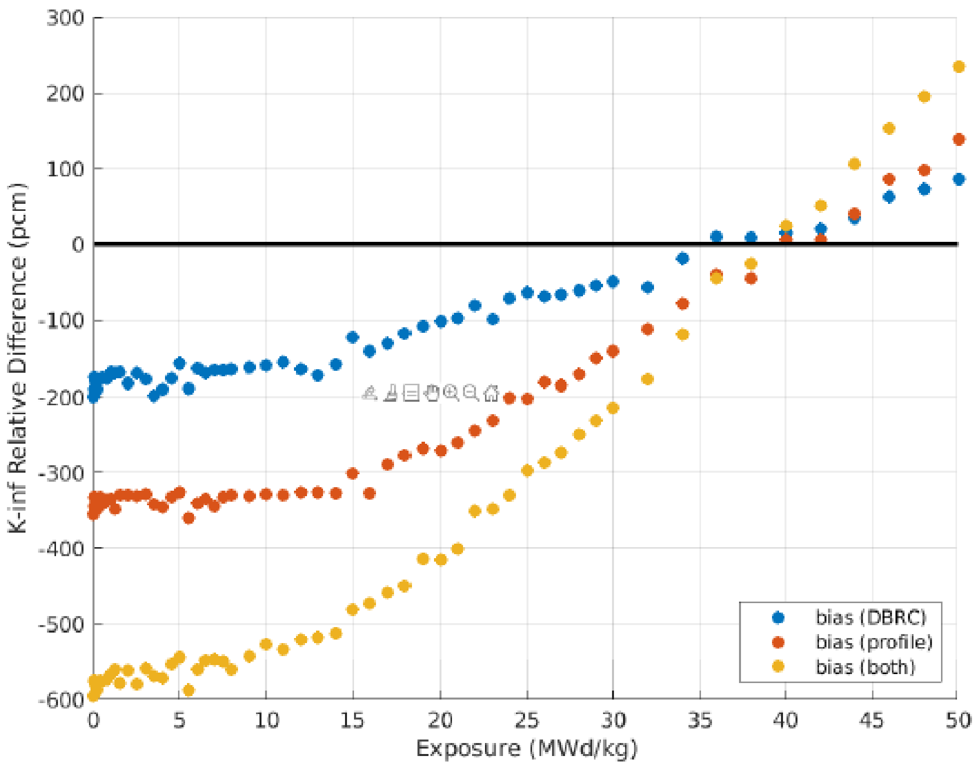
- Comments on the burn-up differences discussed on Day-1 and performed by Andrej and colleagues. Total energy release is overridden by OpenMc to match the values used by Serpent. WIMS library relies on the value reported in the nuclear data library. OpenMc and Serpent used the same Q-values but they changed the value reported by the library.
- Where does energy release (isotope dependent) come from? Many ways to include these values in the code (Serpent) but these were matched to ENDF/B-VII.1.
- It is not clear, but the effect of the energy release changes could be 100-200 pcm.
- Three different ways to define energy release in Serpent: at the end of the burn-up curve these models show differences up to 200 pcm. A different energy release defines a different neutron flux (Oscar's slides).
- What is done in CASMO? Maybe similar to WIMS.
- To define specifications (energy release, DBRC, FPY, ….) used to perform burn-up calculations, and to make comparison among different codes meaningful (Yaron).

3. RECOMMENDATIONS AND ACTIONS

Burnup issues:

JEFF-311 and B71 (which are very close) are the industry reference libraries. They were calculated before DBRC was studied. It is recommended to include DBRC in all calculations with new libraries.

The effect of DBRC (Doppler-Broadened Rejection Correction) was shown (see Hursin presentation) to be about -200 pcm at BOL and +100pcm at EOL (50 GWd/MT). The profile temperature effect may introduce an additional burnup correction. The DBRC correction is almost independent of the analysed system.



It was also found that Erel (total energy release in fission) and the available energy released per fission (fission Q-value) are treated differently in different codes. Some codes do not read the library information but use parameterizations or fixed input values.

Code treatment differences may lead to absolute reactivity differences up to about 200pcm at the EOL. For XS comparison, it is easier to use a constant Erel value independent of the library. However, for the final test we need to use library values, as those will be used by potential users (depending on the code). Additionally, average energy released in capture may induce additional differences.

See the General discussion about "q values in burnup calculations".

The problem of burnup was highlighted in JEFF and CSWEG meetings. It was extensively analysed for ENDF/B libraries by Kang Seog Kim and William A. Wieselquist, "Neutronic Characteristics of ENDF/B-VIII.0 Compared to ENDF/B-VII.1 for Light-Water Reactor Analysis". J. Nucl. Eng. **2**(4) (2021) 318–335, https://doi.org/10.3390/jne2040026. A similar study in JEFF identified ^{239}Pu JEFF-33 as the most important contributor to the burnup differences relative to the JEFF-311 library. However, a significant impact of FPY and XS of FPs was also noted.

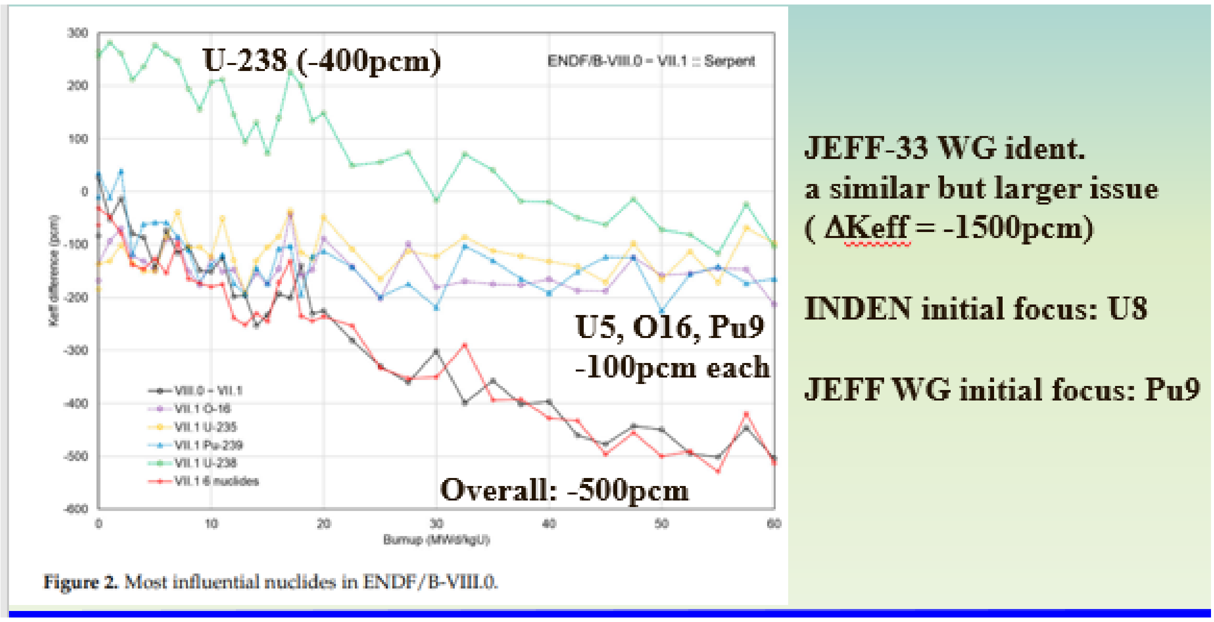


Figure 2. Most influential nuclides in ENDF/B-VIII.0.

In ENDF/B libraries the ^{239}Pu RRR did not change from B71 to B8.0 version, so the impact of ^{239}Pu for B8/B71 burnup differences was small (-100 pcm). The situation changed once the ^{239}Pu thermal PFNS was updated and new ^{239}Pu RPs were developed for INDEN ^{239}Pu files adopted for ENDF/B-VIII.1beta versions.

There are two new ^{240}Pu and ^{241}Pu solutions from JEFF (for each nuclide). One candidate for ^{240}Pu and ^{241}Pu evaluations proposed by Gilles (included in INDEN), the second one by Rochman et al. which were derived by adjustment (which eventually was not used).

There are four new ^{239}Pu solutions (B81b2, B81b2nu1 with 0.3% reduced nubar up to 0.4eV and two JEFF-4T3 candidates – Noguere et al and Rochman et al).

^{239}Pu evaluations were updated considering changes in capture/fission ratio (alpha) and multiplicity, as well as constraints given by the TNC (Standards). The E81b2 (and Noguere et al?) RPs were shown to be in better agreement with preliminary alpha ratio measured at n_TOF than previous evaluations, especially below 10 eV.

These new Pu-isotope evaluations were tested considering PST (as a function of Ea, ATLF), Mistral-2 RTC, and depletion (using different configurations and enrichment – pincell, 3x3, 17x17, etc).

Burnup dependence is listed in tables below. The DBRC correction was not applied to listed results.

	Wemple BOL Studvisk [pcm]	OpenMC(1) JSI [pcm]	Serpent(2) ORNL [pcm]	WIMS JSI [pcm]	WIMS(3) Cabellos [pcm]	Dragon Hursin [pcm]
B8/B71	-150	-130	0	-200	-273	-150
B81b2/B71	-150	-30	-30	-100	-239	-75
B81b2nu1/B71	--	-30	-30	--	-239	??
Rochman /J311						+100
JEFF4T2/J311						-700
JEFF-33/J311						+400

1) Erel was fixed to B80 and not read from the libraries, 4.75% enrichment, 300ppm boron.
2) ^{241}Pu was not included in calculations, 6 nuclides from the paper (^{235}U, O, ^{238}U, ^{240}Pu, ^{242}Pu, ^{239}Pu)
3) 4.8% enrichment, 17x17, 500 ppm boron.

EOL: 60MWd/kgU

	Wemple EOL Studvisk [pcm]	OpenMC(1) JSI [pcm]	Serpent(2) ORNL [pcm]	WIMS JSI [pcm]	WIMS(3) Cabellos [pcm]	Dragon Hursin [pcm]	Reactiv. Swing (EOL-BOL) [pcm]
B8/B71	-450	-250	-450	-500	-600	-475	-325
B81b2/B71	-200	0	-300	-250	-366	-225	-150
B81b2nu1/B71	--	0	-350	--	-401	??	??
Rochman /J311						+100	+200
JEFF4T2/J311						-700	-1000
JEFF-33/J311						-900	-1300

1) Erel was fixed to B80 and not read from the libraries, 4.75% enrichment, 300ppm boron.
2) ^{241}Pu was not included in calculations, 6 nuclides from the paper (^{235}U, O, ^{238}U, ^{240}Pu, ^{242}Pu, ^{239}Pu)
3) 4.8% enrichment, 17x17, 500 ppm boron.

Additionally, the PST benchmarks were used to compare different evaluations.

	PST thermal (C-E) [pcm]
B8/B71	0
B81b2/B71	+500
B81b2nu1/B71	+200
Rochman /J311	>300
JEFF4T22	¿?

^{235}U

Small changes in ^{235}U RRR compared to JEFF-33 and B8 do not impact depletion (about +150pcm for B81b2 set, and zero for JEFF-4T2 ^{235}U set).

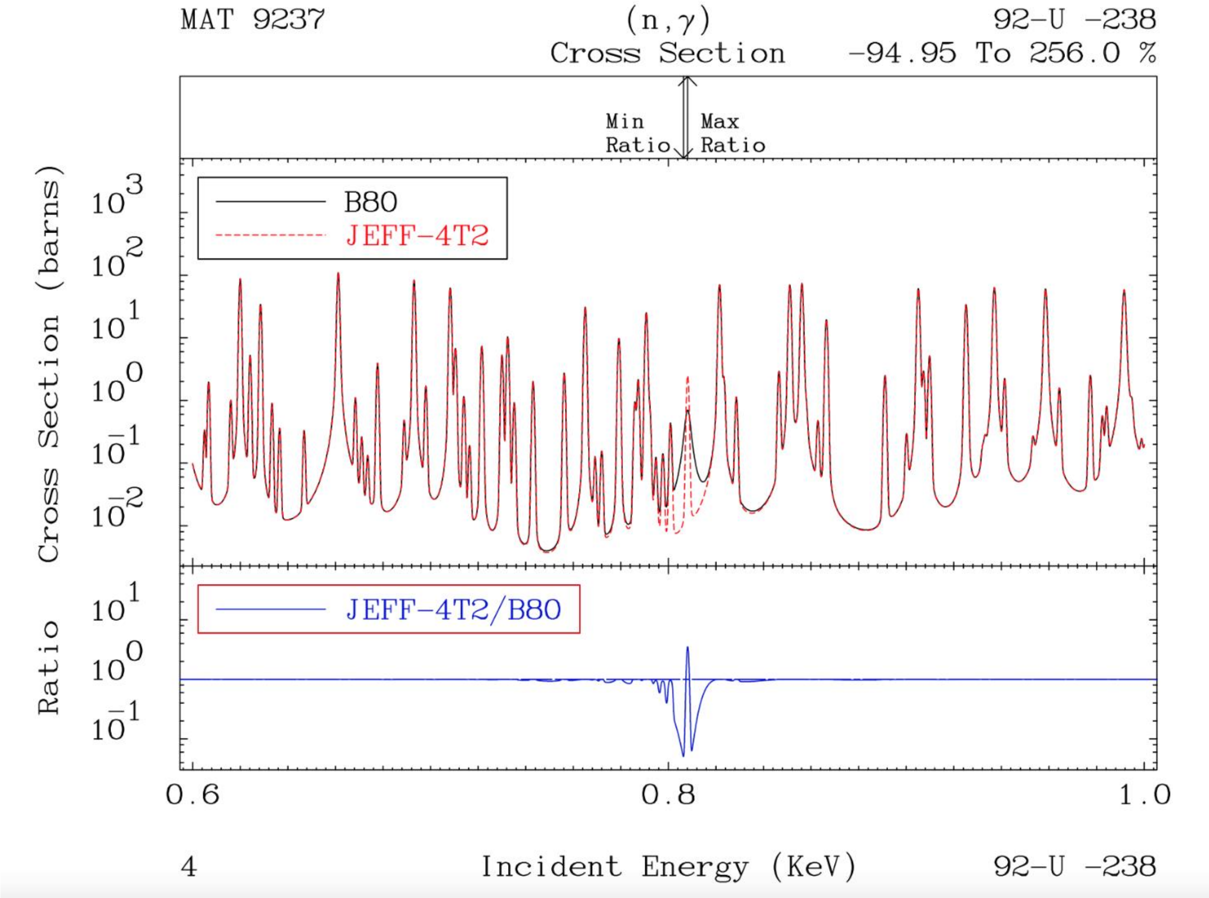
^{238}U

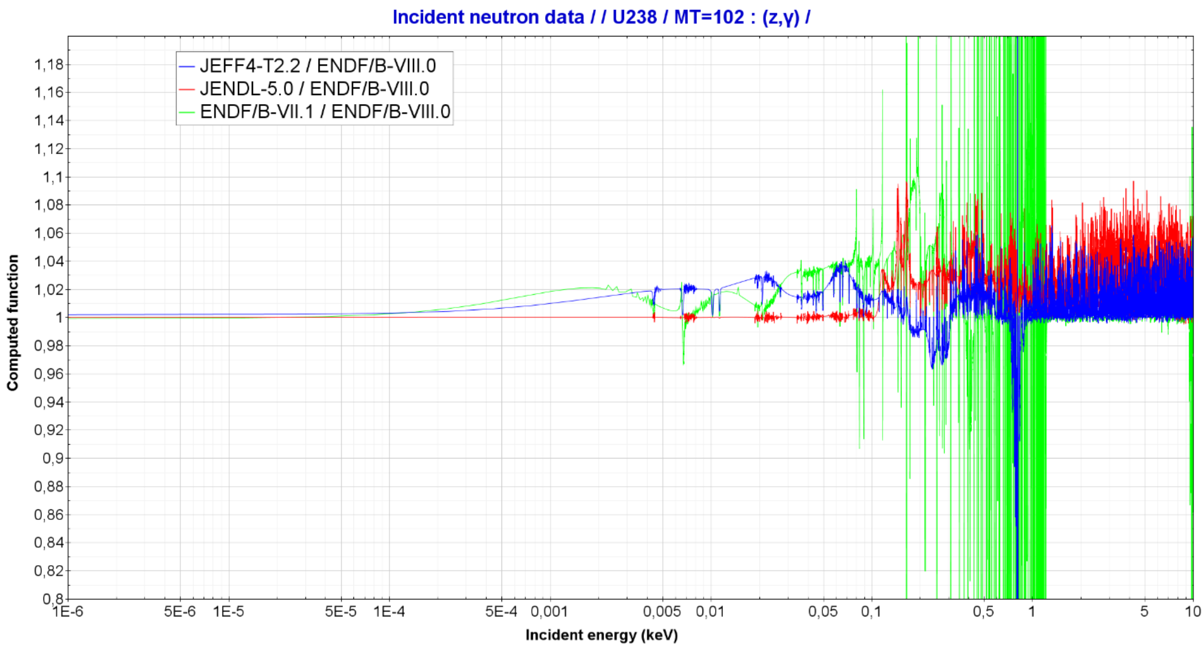
A typo in the ^{238}U RPs for the 808 eV p-wave G_g parameter was found by Cabellos et al [JEFDOC-2097] in ENDF/B-VIII.0 and JENDL-5.0 files:

8.081378+2 5.000000-1 5.333600-4 2.250000**+0** 0.000000+0 0.000000+09237 2151 1036

The '**+0**' should be modified by '**-2**'"

The impact of the typo on cross section is shown in the figure below.



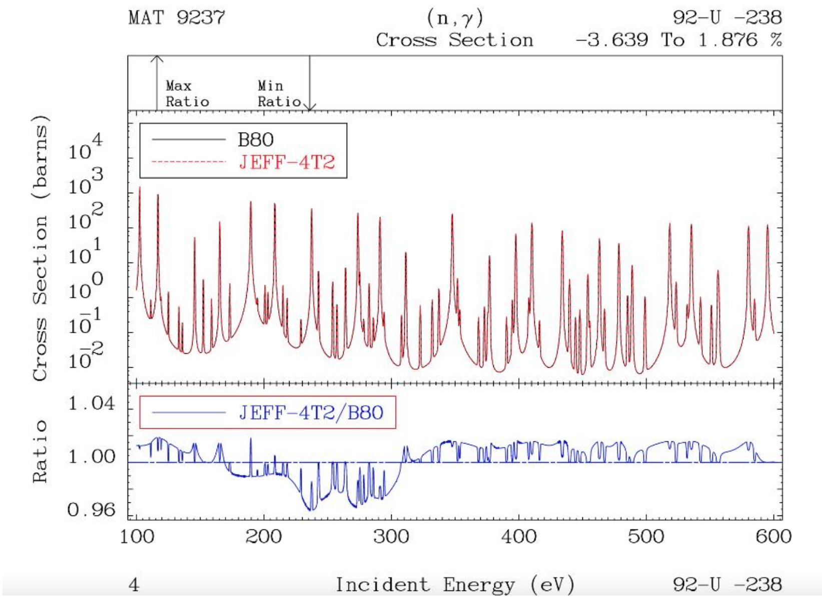


There are two solutions, see the figure above:

1) JEFF-4T2.2 in blue (Kopecky et al) revised ^{238}U RPs as follows:
   - All gamma widths (Gg) for the positive resonances up to 500 eV were increased by 1.6%.
   - The Gg widths for the negative resonances were modified so that the calculated thermal capture cross section is in agreement with the value presented Trkov et al. (Nucl. Sci. Eng. 150 (2005) 336).

     Such modifications correspond to a preliminary work, indicating a possible direction for resolving part of the burnup issue.

     The observed reduction of the capture cross section between 180 and 300 eV should be further investigated. The above-mentioned 1.6% Gg increase is not expected to reduce the capture XS. Bound state impact should be assessed.



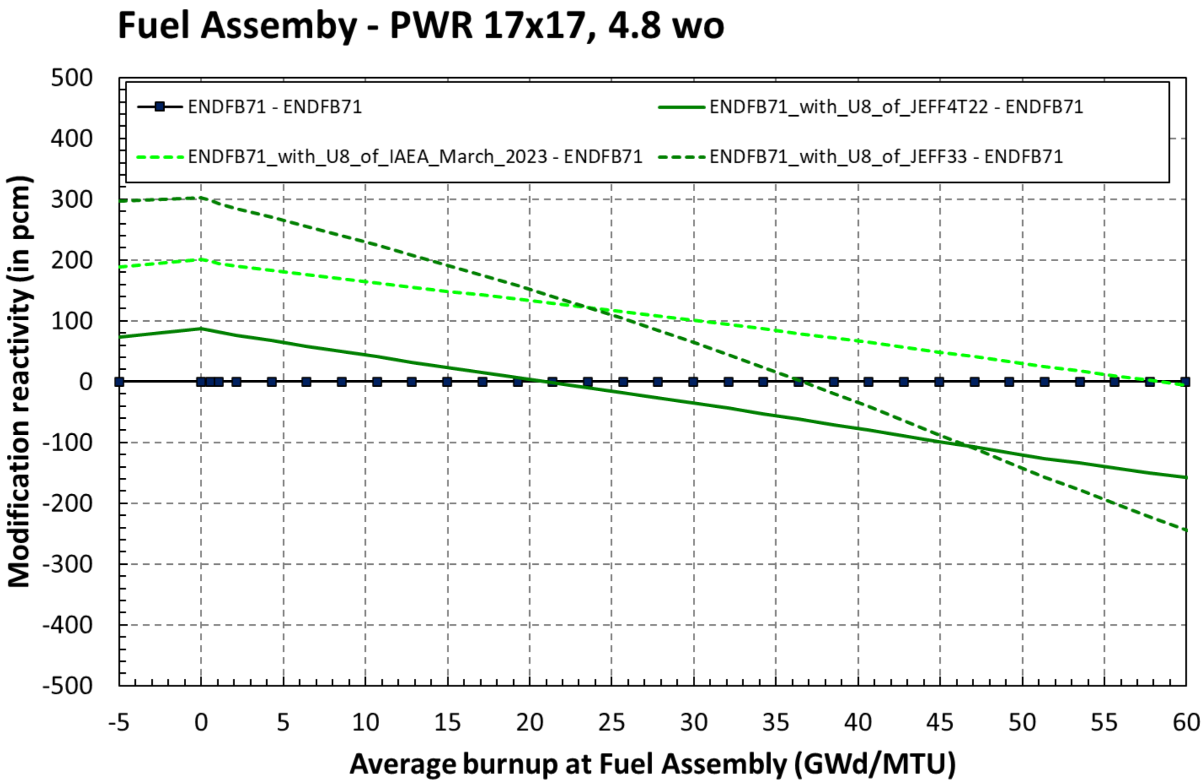
2) JENDL-5 in red, XS were adjusted using fast critical systems (and 6 systems for Na void reactivity), see Iwamoto presentation.

Despite very significant differences in derivation, both ^{238}U RP solutions show a net increase in capture cross sections above 100 eV (with the exception of the 180-300 eV region for Solution #1).

Cabellos calculated the burnup behaviour of those two sets of ^{238}U RPs; results are shown in the figure below compared to the RPs of the ^{238}U IAEA CIELO evaluation adopted by both JEFF-3.3 and ENDF/B-VIII.0 libraries. The IAEA CIELO evaluation overestimated the B71 reactivity at the BOL by 300pcm and resulted in -250 pcm loss of reactivity compared to B71 at assumed EOL (=60 MWd/kgU) for an overall swing of 550pcm.

Solution #1 (JEFF-4T22) shows +100pcm at the BOL and -150pcm at EOL for a reduced swing of 250pcm. Solution #2 (JENDL-5) shows +200 pcm at BOL and 0pcm relative to B71 at EOL for a reduced overall reactivity swing of 200pcm.

Note that these swings will be reduced by about 300 pcm (-200 pcm at BOL and +100pcm at EOL) if the DBRC correction is considered in burnup/depletion calculations of the new libraries.
In summary, we do have two new solutions for ^{238}U RPs that reduce the reactivity swing from BOL to EOL to about +/- 100 pcm (-50pcm for the solution #1 and +100pcm for the solution #2).


Fuel Assemby - PWR 17x17, 4.8 wo
ENDFB71 - ENDFB71
ENDFB71_with_U8_of_JEFF4T22 - ENDFB71
ENDFB71_with_U8_of_IAEA_March_2023 - ENDFB71
ENDFB71_with_U8_of_JEFF33 - ENDFB71
Modification reactivity (in pcm)
500
400
300
200
100
0
-100
-200
-300
-400
-500
-5
0
5
10
15
20
25
30
35
40
45
50
55
60
Average burnup at Fuel Assembly (GWd/MTU)

IAEA Technical Meeting of INDEN on Nuclear Data Evaluation of Fissile Actinides

20 – 23 November 2023
IAEA, Vienna

ADOPTED AGENDA

Monday, 20 November (09:30 am – 05:30 pm, open 09:20 Vienna time)

09:30	**Welcome and introduction,** R. Capote Noy / Scientific Secretary **Election of Chair and Rapporteur(s), adoption of Agenda**	
10:00	**Participants' Presentations** (~60' each w/ discussion)	
	D. Rochman	Adjustment of ^{239,240,241}Pu for the JEFF4T3 library
	O. Cabellos	Status of Burnup calculations
	M. Hursin	A few slides on optimization of RRR parameters
13:00	*Lunch break*	
14:00	**Participants' Presentations** cont' (~60' each w/ discussion)	
	A. Trkov	Loss of reactivity in burnup calculations using recent data libraries
	R. Capote	Overview of CSWEG presentations on burnup issue
	Discussion	*Coffee breaks as needed*

Tuesday, 21 November (09:30 am – 05:30 pm)

10:00	**Participants' Presentations** cont' (~60' each w/ discussion)	
	E. Leal Cidoncha	Measurement of the neutron-induced capture-to-fission cross-section ratio in ^{233}U at LANSCE"
	A. Wallner	Neutron-induced reactions on ^{232}Th, ^{235}U and ^{238}U with thermal and fast neutrons - final results from AMS measurements
	Y. Danon	Validation measurements of neutron capture gamma cascades.
	I. Duran	Actinide evaluations up to 300 MeV
13:00	*Lunch break*	
14:00	**Participants' Presentations** cont' (~60' each w/ discussion)	
	M. Pigni	INDEN evaluations of fissile actinides
	R. Capote	Comments on INDEN evaluations – nubar & PFNS
	Discussion	*Coffee breaks as needed*

19:00 Dinner at a restaurant (separate information)

Wednesday, 22 November (09:30 am – 05:30 pm)

09:30	**Technical discussions and drafting of the meeting summary report**
13:00	*Lunch break*
14:00	**Technical discussions and drafting of the meeting summary report**
	Coffee breaks as needed

Thursday, 23 November (09:30 am – 3:00 pm)

09:30	**Technical discussions and drafting of the meeting summary report (continue)**
13:00	**Closing of the meeting**
	Coffee and lunch break as needed

ANNEX II

IAEA Technical Meeting of the International Nuclear Data Evaluation Network on Actinide Evaluation in the Resonance Region

20-23 November 2023, IAEA, Vienna (hybrid)

PARTICIPANTS

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ANNEX III

Participants' presentations

#	Author	Title	Link
1	O. Cabellos	Status of Burnup calculations: ENDF/B VIII. 1b2	PDF
2	R. Capote	Actinide evaluations below 100 keV: toward ENDF/B-VIII.1	PDF
3	Y. Danon	Validation Measurements of Neutron Capture Gamma Cascades	PDF
4	I. Duran	Evaluating fission of actinides from 1 MeV to 300 MeV	PDF
5	I. Duran	New Integral references for neutron-induced fission reactions in the RRRs	PDF
6	E. Leal Cidoncha	Measurement of the neutron-induced capture-to-fission cross section ratio in ^{233}U at LANSCE	PDF
7	M. Hursin	Testing new NDLs	PDF
8	M. Pigni	ORNL contributions to the ENDF/B-VIII.1 library	PDF
9	D. Rochman	Adjustment of ^{239,240,241}Pu for the JEFF$_4$T$_3$ library	PDF

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