WPEC Subgroup Proposal

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Title

U-235 Capture Cross Section in the Energy Region from 100 eV to 1 MeV

Short Justification for a Subgroup

The cross sections of ^{235}U were examined by WPEC Subgroup 18 focusing on the thermal and epithermal energy region. As a result, the ORNL group evaluated resolved resonance parameters of ^{235}U up to 2.25 keV, which are currently used for all major libraries. Using the resolved resonance parameters, prediction accuracy of neutronic characteristics is satisfactory for thermal reactors. However, recent benchmark analyses have revealed a problem regarding the capture cross sections of ^{235}U for fast-neuron critical experiments BFS and FCA using U fuels. The existing libraries JENDL-3.3, ENDF/B-VII.0 and JEFF-3.1 cannot describe the sodium voided reactivity and criticalities for these assemblies. The problem is attributable to large capture cross sections of ^{235}U in the energy region form 100 eV to 2.25 keV where the same resolved resonance parameters are used for major libraries. Moreover, in the energy region from 30 keV to 1 MeV, there exists a big difference of ^{235}U capture cross sections between JENDL-3.3 and ENDF/B-VII.0.

The purpose of the subgroup is to address the problem of the U-235 capture cross section from the viewpoints of differential and integral data analyses and then obtain recommended cross sections in the energy region from 100 eV to 1 MeV.

Subgroup Coordinator

JENDL: O. Iwamoto

Subgroup Monitor

JEFF or ENDF

List of Subgroup Participants

JENDL: O. Iwamoto, T. Nakagawa, G. Chiba, S. Okajima, M. Ishikawa (JAEA)
JEFF: R. Jacqmin (CEA), ...
ENDF: R. McKnight (ANL), L.C. Leal (ORNL), T. Kawano (LANL), C. Lubitz (KAPL) ....

Definition of the Project

The project will investigate the problems seen for the BFS and FCA-IX critical experiments using U fuels. Sensitivity analyses of integral data are performed with respect to JENDL-3.3, JEFF-3.1, and ENDF/B-VII.0. If experiments on fast-neutron cores with U fuels are available other than BFS and FCA-IX, they should be included in the benchmark analyses. In parallel, the capture cross section of ^{235}U is reviewed on the basis of differential data. Combining integral and differential data analyses, the project focuses on the cross section and the energy region which give large impacts on the poor prediction of neutronic characteristics for BFS and FCA-IX. Then, the cross sections are re-evaluated as well as resonance parameters.

Using the re-evaluated cross sections and resonance parameters, the BFS and FCA-IX experiments are re-analyzed. Moreover, JAEA is planning to carry out sodium-voided reactivity experiments with U fuels at FCA. The new integral data help to validate the re-evaluated data.

Justification of the Project

Over the years, great efforts have been done to obtain reliable cross sections of ^{235}U, which are the most important physical constants in the nuclear energy applications. The epithermal capture cross section of ^{235}U was investigated within the framework of WPEC Subgroup 18. As a result, the ORNL group revised the resolved resonance parameters, which are currently used by all major libraries. However, recent analyses of fast-neutron cores with U fuels reveal very poor predictions of neutronic characteristics. The C/E values of sodium-voided reactivity for BFS are much less than 1.0 except for JENDL-3.2 which adopted different evaluations. Moreover, the C/E values of criticalities for FCA IX-series experiments, where neutron spectra were varied with U fuels, largely depend on the spectrum shape. From the sensitivity analyses, it has been found that these problems arise more or less from too large capture cross sections of ^{235}U in the energy region from 100 eV to around 2 keV.

There is a big difference between JENDL-3.3 and ENDF/B-VII.0 ^{235}U capture cross sections in the energy region from 30 keV to 1 MeV. The criticalities for the BFS experiments are almost reproduced by ENDF/B-VII.0 which adopted smaller capture cross sections of ^{235}U than JENDL-3.3, while the JENDL-3.3 calculations underestimate the criticalities. However, the ^{235}U capture cross section in ENDF/B-

VII.0 seems too small as compared with available experimental data. Therefore, in addition to the lower energy region from 100 eV to 2.25 keV, we should also examine the data in the higher energy region from 30 keV to 1 MeV.

The upper boundary of the resolved resonance parameters evaluated by the ORNL group is 2.25 keV. This means that part of the resolved resonance region is involved in the problems mentioned above. The new subgroup deals with the issues both from differential and integral sides. In the preliminary analysis at JAEA, energy-averaged capture cross sections of ^{235}U calculated from the resolved resonance parameters evaluated by the ORNL group obviously show overestimate of available experimental data. Further detailed investigation should be done in the new Subgroup.

Relevance to Evaluated Data Files

The problems mentioned can be seen for all major libraries, and must be of interest to all evaluated file projects.

Deliverables

- Recommendation for the capture cross section of ^{235}U in the energy region from 100 eV to 1 MeV.
- Results of benchmark analyses including the new FCA sodium-voided reactivity experiments with U fuels.

Time-Schedule and Milestones

- 1st year: Re-evaluation of ^{235}U capture cross sections, re-analysis of resonance parameters and survey of available benchmarks.
- 2nd year: Start up of sodium-voided reactivity experiments at FCA. Benchmark calculations and feedback to evaluations.
- 3rd year: Recommendation and preparing a final report.

Appendix 1

Problem of capture cross section of U-235 in integral data analyses

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It has been shown that the existing libraries JENDL-3.3, ENDF/B-VII.0 and JEFF-3.1 cannot describe the sodium voided reactivity (SVR) and criticalities for fast-neutron critical experiments BFS and FCA using UO_2 fuels. The present report reviews this problem briefly.

Figure 1 shows C/E values of SVR measured at BFS-62-2 and BFS-62-3A. Calculated

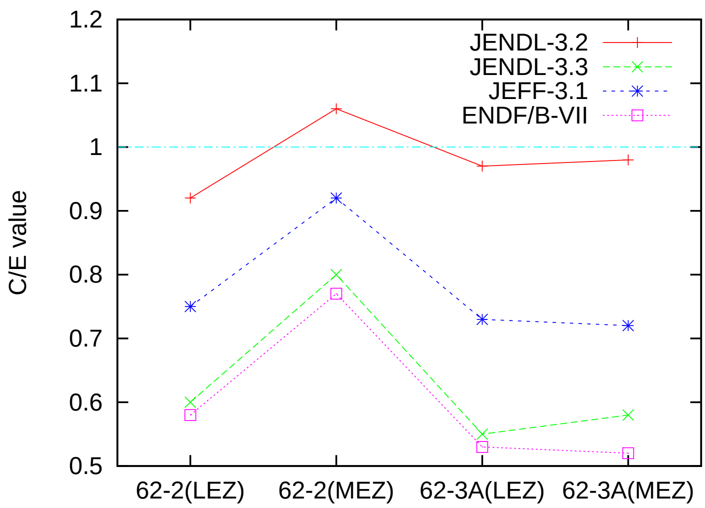

Figure 1: C/E values of sodium voided reactivity (LEZ: Low enriched U zone, MEZ: Medium enriched U zone)

values with JENDL-3.2 agree well with experimental values while other libraries significantly underestimate. The difference in C/E values between JENDL-3.2 and other libraries are caused mainly by differences in capture cross section of U-235 below 2.5 keV.

At FCA, three assemblies, IX-1, IX-2 and IX-3, were constructed without MOX fuels. Neutron spectra of these assemblies were changed with adjusting amount of diluents material, carbon. The IX-3 assembly has the hardest neutron spectrum. **Figure 2** shows C/E values of criticality. C/E values obtained with all the libraries except for JENDL-3.2 strongly depend on assemblies (hardness of neutron spectrum). On the other hand, the dependence is not observed in the results obtained with JENDL-3.2. It was shown that the sensitivity of criticality to capture cross section of U-235 above 30 keV is almost same in these three assemblies. Hence, it can be said that this dependence of C/E values are strongly related to capture cross sections of U-235 below 2.5 keV, and this result supports the evaluation of JENDL-3.2 for U-235 capture cross sections below 2.5 keV as the BFS integral analyses.

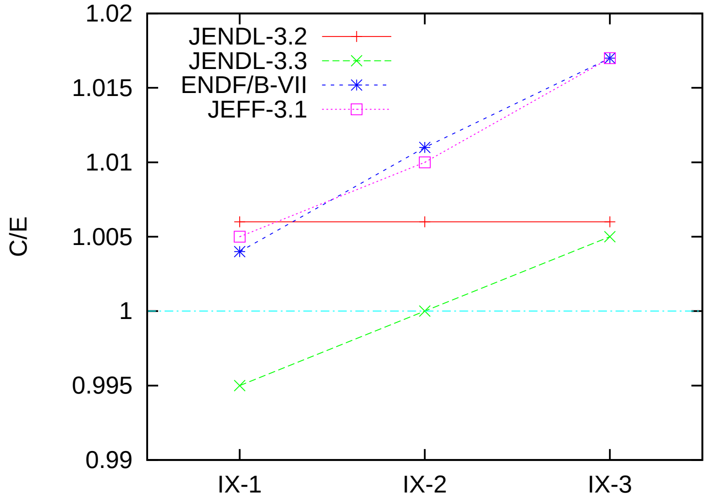

Figure 2: C/E values of criticality of FCA IX assemblies

C/E values for criticality of BFS are shown in **Fig.3**. These cores have different sensitivity to cross sections of U-235 (the 62-1 core is the most sensitive to U-235). All the libraries except

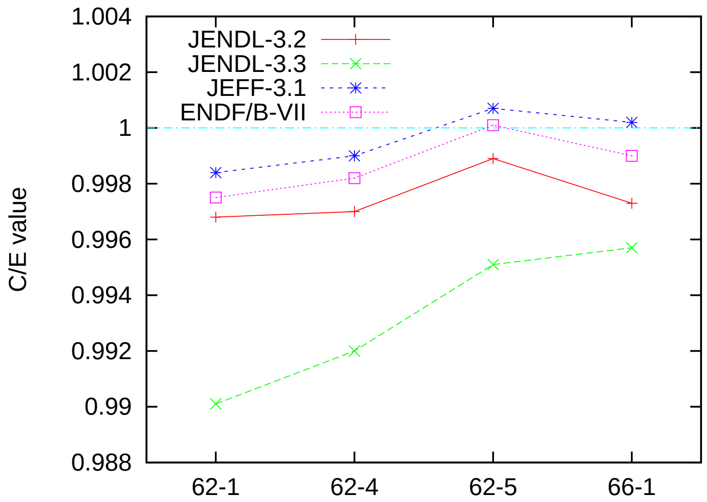

Figure 3: C/E values of criticality

for JENDL-3.3 predicts well. The results of the afore mentioned integral analyses suggest that capture cross section of U-235 below 2.5 keV seems to be too large. When we replace capture cross section of U-235 in ENDF/B-VII or JEFF-3.1 by that of JENDL-3.2 below 2.5 keV, these criticalities will be overestimated about 0.5%$\Delta k/kk'$ by ENDF/B-VII or JEFF-3.1. The reason why JENDL-3.2 with small cross sections of U-235 below 2.5 keV predicts these criticalities well is that capture cross section of U-235 above 30 keV is larger than that of ENDF/B-VII or JEFF-3.1. This difference above several tens of keV also affects the 'absolute trend' of the C/E values of FCA IX cores (ENDF/B-VII and JEFF-3.1 systematically overestimate these criticalities). Hence, capture cross section of U-235 above 30 keV should be taken into consideration when this cross section below 2.5 keV is discussed.

Appendix 2

^{235}U Capture Cross Section

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Capture cross section in the resonance region

Currently available evaluated nuclear data files represent the ^{235}U capture cross section in the form of resonance parameters which were evaluated by Leal et al. [Le99] According to the paper of Leal et al., they made shape analysis of various experimental data with the SAMMY code. The experimental data on the capture cross section considered in the analysis were those of Perez et al. [Pe73] in the energy range up to 100 eV, and de Saussure et al.[De67] up to 2.25 keV.

Figure 1 compares the average capture cross sections calculated from the resonance parameters with the average data of Perez et al. and de Saussure et al. Only JENDL-3.3 is shown in the figure because the ENDF/B-VII.0 and JEFF-3.1 are the same as JENDL-3.3 below 2.25 keV. It is clearly seen that the calculated cross sections from JENDL-3.3 are in good agreement with the data of de Saussure et al. in the energy region up to about 1.4 keV, and they are larger than the experimental data above 1.4 keV. The data of Perez et al. are quite smaller than JENDL-3.3.

The capture cross section is measured frequently in the form of α (the ratio to the fission cross section). The α values calculated from JENDL-3.3 are shown in **Fig. 2** together with experimental data. Average α values strongly depend on a derivation method. The average values of JENDL-3.3 in the figure are the ratios of average capture cross sections and average fission cross sections. Therefore experimental data obtained by averaging pointwise α values measured are excluded from the figure. JENDL-3.3 is larger than the experimental data in the resolved resonance region. Since the fission cross section of ^{235}U has only small uncertainties, we can suppose the capture cross sections of de Saussure et al. seem to be too large by comparing Figs. 1 and 2. The average α of ENDF/B-VI.2 is also given in the figure. It is too small below 500 eV but quite reasonable above 500 eV.

Capture widths given in JENDL-3.3 are shown in **Figure 3**. Leal et al. reported the average capture width of 38.03±1.70 meV from 89 resonances and 38.14±1.70 meV from 35 strong resonances in the energy range from thermal to 50 eV. Mughabghab [Mu06] recommended the capture width of 38.1±1.7 meV. The capture widths given in the current evaluated data files are larger than these average values and scattered resonance by resonance. The simple average of these capture widths is 43.3 meV which is 14% larger than 38.1 meV. A dark band in the figure might be initial values for the SAMMY analysis which have not been changed largely after SAMMY run. These initial values are larger than 38.1 meV.

From the above considerations, we can point out that the present ^{235}U resolved resonance parameters stored in JENDL-3.3, ENDF/B-VII and JEFF-3.1 yield possibly too large capture cross sections.

Capture cross section in the smooth region

Figure 4 shows the capture cross section above the resolved resonance region. Large discrepancies are found among evaluated data sets. We tried data analyses with the GMA code in this energy region. New results are smaller than JENDL-3.3 and however larger than ENDF/B-VII.0. These discrepancies must be solved because the ^{235}U capture cross section is important for reactor applications.

References

[Le99] L.C.Leal et al., *Nucl. Sci. Eng.*, **131**, 230 (1999).
[De67] G.de Saussure et al., ORNL/TM-1804 (1967).
[Pe73] R.B.Perez et al., *Nucl. Sci. Eng.*, **52**, 46 (1973).
[Mu06] S.F.Mughabghab, *Atlas of Neutron Resonances*, Elsevier (2006).

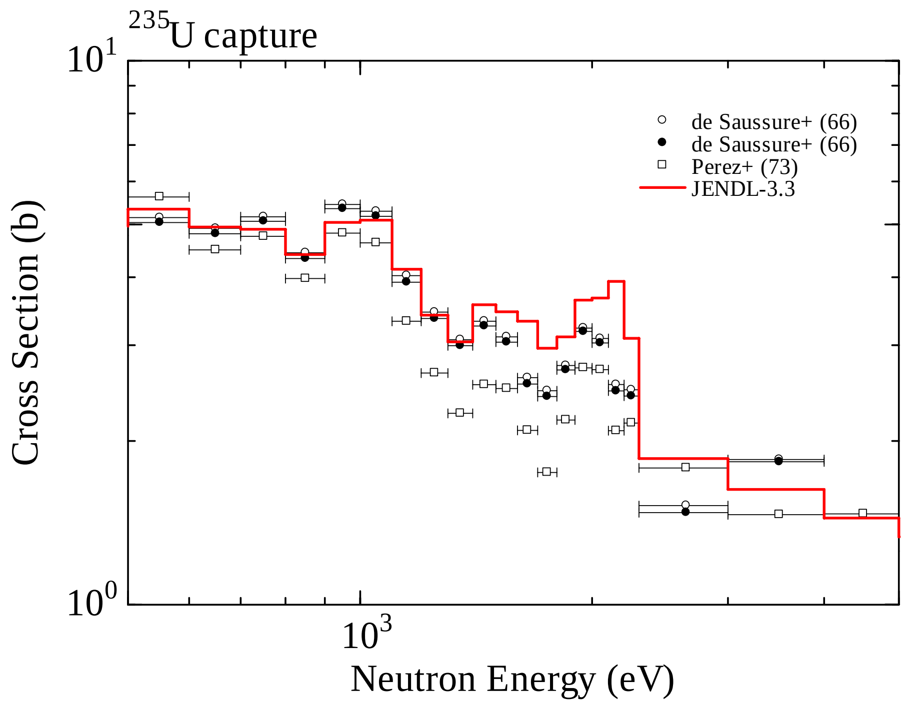

Fig. 1 ^{235}U capture cross section

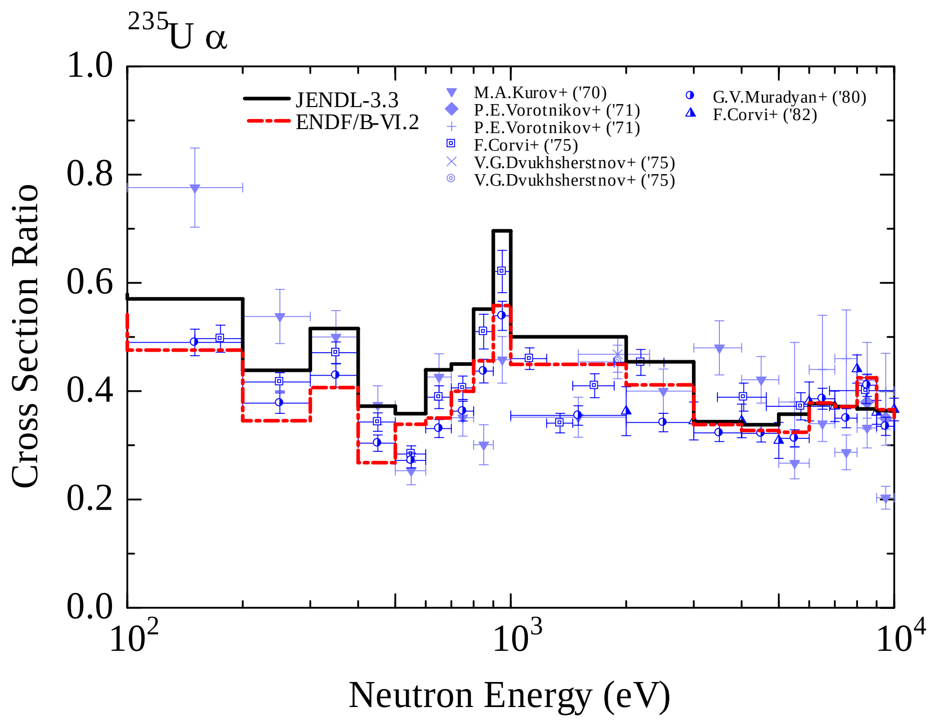

Fig.2 ^{235}U capture to fission ratio (α)

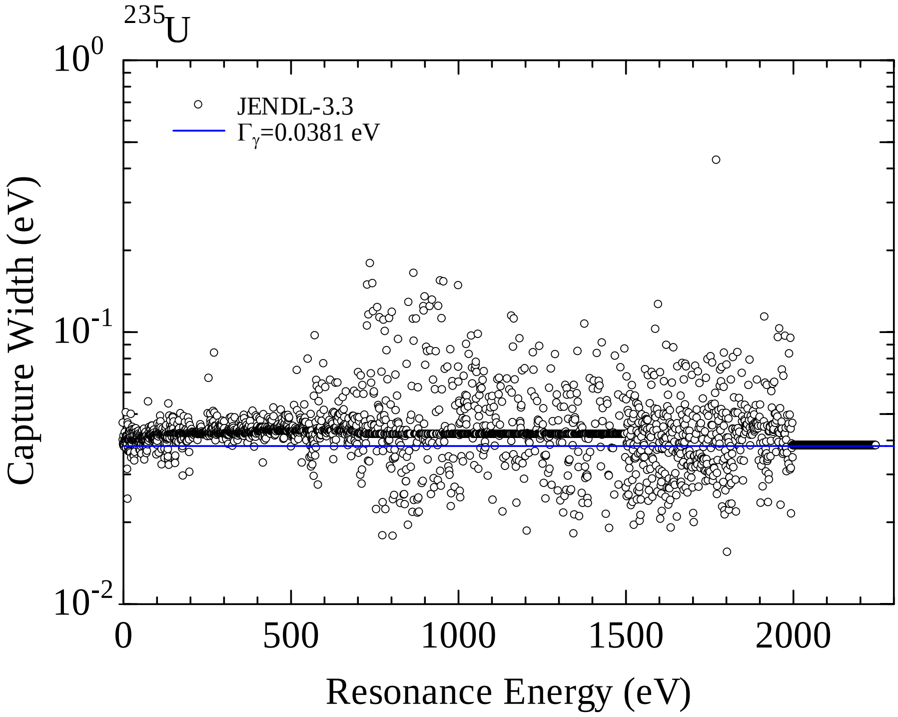

Fig. 3 Capture widths of ^{235}U resonances

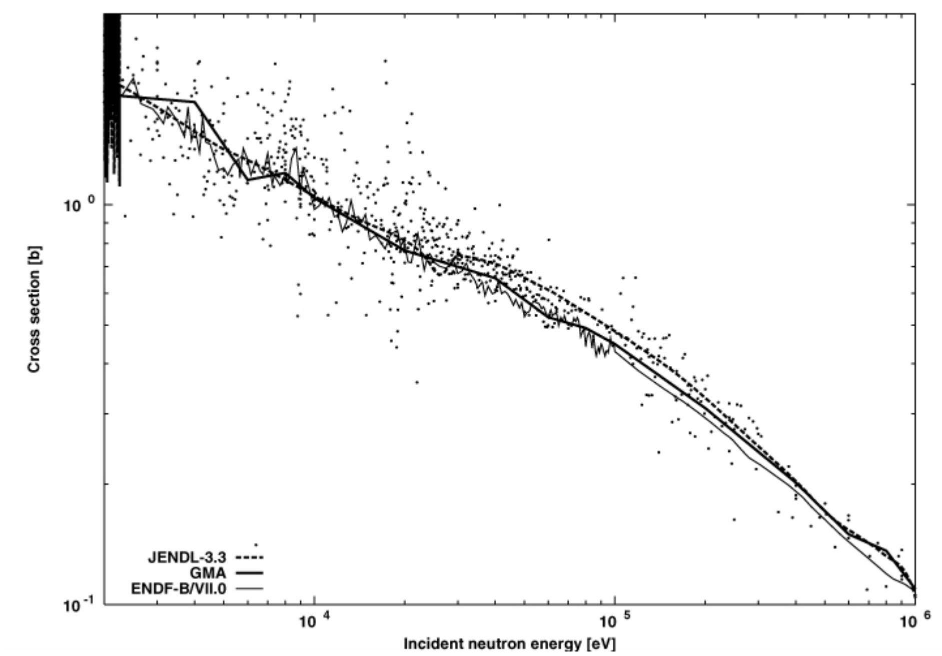

Fig. 4 ^{235}U capture cross section (above 2 keV)