Meeting Nuclear Data Needs for Advanced Reactor Systems
Working Party on International Nuclear Data Evaluation Co-operation

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A report by the Working Party on International Nuclear Data Evaluation
Co-operation of the NEA Nuclear Science Committee

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Foreword

The Working Party on International Nuclear Data Evaluation Co-operation (WPEC) has been established under the aegis of the OECD/NEA Nuclear Science Committee (NSC) to promote the exchange of information on nuclear data evaluations, validation and related topics. Its aim is also to provide a framework for co-operative activities between the members of the major nuclear data evaluation projects. This includes the possible exchange of scientists in order to encourage co-operation. Requirements for experimental data resulting from this activity are compiled. The WPEC determines common criteria for evaluated nuclear data files with a view to assessing and improving the quality and completeness of evaluated data.

The parties to the project are: BROND (Russian Federation), ENDF (United States), JENDL (Japan) and JEFF (other NEA Data Bank member countries). Co-operation with evaluation projects of non-NEA countries, especially the Chinese CENDL project, is organised through the Nuclear Data Section of the International Atomic Energy Agency (IAEA).

This report has been issued by the WPEC Subgroup 31, whose mission was to utilise the collective knowledge of the international nuclear data measurement community to consider the appropriate resources to address and meet the data needs quantified by WPEC Subgroup 26 for Advanced Reactor Systems. The members of Subgroup 31 performed reviews of uncertainty evaluations by evaluators, of state-of-art experimental techniques, of current experimental situations, and summarised an appropriate path to meet the requirements.

The opinions expressed in this report are those of the authors only and do not necessarily represent the position of any member country or international organisation.
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1. Introduction

The nuclear data needs for advanced reactor systems were identified by the activities of WPEC Subgroup 26 (SG26) in the report on "Uncertainty and Target Accuracy Assessment for Innovative Systems Using Recent Covariance Data Evaluations" [1]. The target accuracy assessment was also performed in order to provide quantitatively improvement requirements of nuclear data by isotope, nuclear reaction and energy range [1]. It was systematically shown that there are significant gaps between the current uncertainties and the target accuracies. To find correct paths to reduce the gaps and find correct paths to meet these nuclear data needs, the WPEC Subgroup 31 “Meeting Nuclear Data Needs for Advanced Reactor Systems” was organised. To utilise the collective knowledge of the international nuclear data measurement and evaluation community, the members of the Subgroup 31 (SG31) were comprised of nuclear data measurement experts, together with evaluation experts.

The uncertainties of nuclear data have been investigated and updated independently under different nuclear data evaluation projects, such as JENDL, JEFF, ENDF, BROND, and MANREAD. Since the accuracy requirements and priorities summarised in SG26 strongly depends on the assumed initial uncertainty data, the current uncertainty evaluations in JENDL, JEFF, ENDF, BROND, and MANREAD are compared with those of SG26 in Section 2 for some selected data. These comparisons recognise the existence of a discrepancy on accuracy estimation between different evaluations. In section 2.1, the outline of SG26 report are briefly reviewed together with summaries of target accuracies of nuclear data for each reactor type, such as Pressurized Water Reactors (PWR), Very High Temperature Reactors (VHTR), Fast Reactors (FR) and Accelerator Driven Minor Actinide Burners (ADMAB). In section 2.2, the method of uncertainty evaluation in JENDL-4.0 is reviewed as an example to show current evaluation status in resonance region, unresolved resonance region, calculations using CCONE, and neutron emission multiplicity.

The current status of experimental techniques is reviewed in Section 3. Especially, measurement methods on capture cross sections were described in detail in Section 3.1, where noticeable advancements on measurement techniques and also possible origins on main uncertainty were identified. Experimentalist’s reviews on measurements are also given on capture cross-section for $^{28}$Si, $^{206}$Pb, $^{238}$U, $^{241,243}$Am and $^{237}$Np. The current status on the $^{10}$B(n,$\alpha$) reaction and on inelastic scattering measurements are briefly described in Sections 3.2 and 3.3, respectively. In Section 3.4, recent progress in fission experiments using inverse kinematics is summarised. Although this is an indirect measurement method, it is expected to have important contribution because the method can extend the diversity of fissioning systems.

Based on the reviews and discussions on uncertainty evaluations and recent experimental status, collaborative paths towards meeting the needs are discussed and recommended in Section 4. A summary is given in Section 5.

The appendix includes the Graphical Summary of Available Measured Data. This appendix graphically shows available data with their energy range and uncertainty. This gives comprehensive information on experimental data together with their uncertainties.
2. Review of uncertainty evaluation

There are various advancements in developing the methodology of covariance evaluation for each nuclear data project, such as WPEC/SG26, JENDL-4.0, JEFF-3.1.2, ENDF/B-VII.1, BROND-3 and MANREAD.

Each nuclear data project supplied their evaluated uncertainties for selected nuclear data for SG31. The evaluated uncertainties were compared for some selected nuclear data in Table 1. Although almost all evaluated uncertainties agree within a factor of two, there are exceptions as an example of inelastic cross-section for \(^{56}\)Fe. Two evaluation methods (GMA and SYS) in BROND-3 explain half of the gap. However, JENDL-4.0 gives three times larger uncertainties even compared to BROND-3 (SYS). These discrepancies could be partly explained by the limited number of experiments for each nuclear data, and furthermore there are discrepancies between existing experimental data in many cases; it makes statistical treatment difficult. Evaluators need to assume some values for uncertainties based on their own experiences and insights.

Table 1: Comparison of accuracy estimation (%) between SG26, JENDL-4.0, ENDF/B-VII.1, JEFF-3.1.2 and BROND-3

<table>
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<th>Quantity</th>
<th>Energy range</th>
<th>SG26</th>
<th>JENDL-4</th>
<th>ENDF/B-VII.1</th>
<th>JEFF-3.1.2</th>
<th>BROND-3 (GMA)</th>
<th>BROND-3 (SYS)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Na-23</td>
<td>(\sigma_{\text{inel}})</td>
<td>0.498 - 2.23 MeV</td>
<td>25</td>
<td>17 · 20</td>
<td>8 · 16</td>
<td>~ 5</td>
<td>7 · 8</td>
<td></td>
</tr>
<tr>
<td>Si-28</td>
<td>(\sigma_{\text{inel}})</td>
<td>1.35 - 6.07 MeV</td>
<td>14</td>
<td>50</td>
<td>2 · 4</td>
<td>~ 5</td>
<td>2 · 5</td>
<td>2 · 5</td>
</tr>
<tr>
<td>Si-28</td>
<td>(\sigma_{\text{capt}})</td>
<td>6.07 - 19.6 MeV</td>
<td>53</td>
<td></td>
<td>~ 35</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fe-56</td>
<td>(\sigma_{\text{inel}})</td>
<td>0.498 - 2.23 MeV</td>
<td>16</td>
<td>25</td>
<td>15 · 17</td>
<td>7 · 12</td>
<td>2 · 5</td>
<td>5</td>
</tr>
<tr>
<td>Pb-206</td>
<td>(\sigma_{\text{inel}})</td>
<td>1.35 - 2.23 MeV</td>
<td>14</td>
<td></td>
<td>8</td>
<td></td>
<td>10</td>
<td></td>
</tr>
<tr>
<td>Pb-207</td>
<td>(\sigma_{\text{inel}})</td>
<td>0.498 - 2.23 MeV</td>
<td>11</td>
<td></td>
<td>9 · 17</td>
<td></td>
<td>10</td>
<td></td>
</tr>
<tr>
<td>U-238</td>
<td>(\sigma_{\text{inel}})</td>
<td>0.498 - 6.07 MeV</td>
<td>10</td>
<td>20</td>
<td>5 · 11</td>
<td>12 · 21</td>
<td>7</td>
<td></td>
</tr>
<tr>
<td>U-238</td>
<td>(\sigma_{\text{capt}})</td>
<td>2.03 - 24.8 keV</td>
<td>3.9</td>
<td></td>
<td>7 · 10</td>
<td>3</td>
<td>0.9 · 2.2</td>
<td>7</td>
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<tr>
<td>Pu-238</td>
<td>(\sigma_{\text{fiss}})</td>
<td>0.183 - 1.35 MeV</td>
<td>20</td>
<td></td>
<td>2.4 · 3.0</td>
<td>1.0 · 1.2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pu-239</td>
<td>(\sigma_{\text{fiss}})</td>
<td>2.03 - 498 keV</td>
<td>7</td>
<td>15</td>
<td>5 · 9</td>
<td>7 · 15</td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>Pu-240</td>
<td>(\sigma_{\text{fiss}})</td>
<td>0.498 - 1.35 MeV</td>
<td>6</td>
<td></td>
<td>0.8</td>
<td>0.9</td>
<td>1.4</td>
<td></td>
</tr>
<tr>
<td>Pu-240</td>
<td>(\nu)</td>
<td>0.498 - 1.35 MeV</td>
<td>4</td>
<td></td>
<td>0.2</td>
<td>0.4</td>
<td>0.7</td>
<td></td>
</tr>
<tr>
<td>Pu-241</td>
<td>(\sigma_{\text{fiss}})</td>
<td>454 eV - 1.35 MeV</td>
<td>8</td>
<td>20</td>
<td>0.8 · 5</td>
<td></td>
<td>4</td>
<td></td>
</tr>
<tr>
<td>Pu-242</td>
<td>(\sigma_{\text{fiss}})</td>
<td>0.498 - 2.23 MeV</td>
<td>19</td>
<td>21</td>
<td>2</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Am-241</td>
<td>(\sigma_{\text{fiss}})</td>
<td>2.23 - 6.07 MeV</td>
<td>12</td>
<td></td>
<td>1.4</td>
<td>1.3</td>
<td>1.5</td>
<td></td>
</tr>
<tr>
<td>Am-242m</td>
<td>(\sigma_{\text{fiss}})</td>
<td>67.4 keV - 1.35 MeV</td>
<td>17</td>
<td></td>
<td>3</td>
<td>1.5</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cm-244</td>
<td>(\sigma_{\text{fiss}})</td>
<td>0.498 - 1.35 MeV</td>
<td>50</td>
<td></td>
<td>4</td>
<td>3</td>
<td>3 · 7</td>
<td></td>
</tr>
<tr>
<td>Cm-245</td>
<td>(\sigma_{\text{fiss}})</td>
<td>67.4 - 183 keV</td>
<td>47</td>
<td>5</td>
<td>4</td>
<td></td>
<td>3</td>
<td></td>
</tr>
</tbody>
</table>
In Section 2.1, the outline of SG26 report is briefly reviewed together with the summaries of target accuracies of nuclear data for each reactor type, such as Pressurised Water Reactors (PWR), Very High Temperature Reactors (VHTR), Fast Reactors (FR), Accelerator-Driven Minor Actinide Burners (ADMAB).

In Section 2.2, the status of evaluations in JENDL-4.0 is briefly reviewed for the resonance region, unresolved resonance region, calculation using CCODE, and neutron emission multiplicity.

### 2.1 Method of SG26 evaluation

The Working Party on International Nuclear Data Evaluation Co-operation (WPEC) established Subgroup 26 to develop a systematic approach to define data needs for advanced reactor systems and to make a comprehensive study of such needs for Generation-IV (Gen-IV) reactors. The subgroup was established at the end of 2005 [1].

A comprehensive sensitivity and uncertainty study was performed to evaluate the impact of neutron cross-section uncertainty on the most significant integral parameters related to the core and fuel cycle of a wide range of innovative systems, even beyond the Gen-IV range of systems. Results were obtained for the Advanced Breeder Test Reactor (ABTR), the Sodium-cooled Fast Reactor (SFR), the European Fast Reactor (EFR), the Gas-cooled Fast Reactor (GFR), the Lead-cooled Fast Reactor (LFR), the Accelerator-Driven Minor Actinide Burner (ADMAB), the Very High Temperature Reactor (VHTR) and the Pressurised Water Reactor with extended burn-up (PWR). These systems corresponded to current studies in the Generation-IV initiative, the Global Nuclear Energy Partnership (GNEP), the Advanced Fuel Cycle Initiative (AFCI), and in advanced fuel cycle and Partitioning/Transmutation studies in Japan and Europe.

The covariance data used in the uncertainty analysis was the BOLNA [2] matrix developed in a joint effort of several laboratories contributing to Subgroup 26. The analysis was mostly focused on integral parameter uncertainties due to neutron cross-section uncertainties. The integral parameters considered were related to the reactor core performances ($k_{efr}$, reactivity coefficients, power distributions etc.) but also to some important fuel cycle-related parameters, such as the transmutation potential, the doses in a waste repository or the neutron source at fuel fabrication.

Preliminary “Design Target Accuracies” were compiled together and a target accuracy assessment was performed in order to provide an indicative quantitative evaluation of nuclear data improvement requirements by isotope, nuclear reaction and energy range, which would meet the design target accuracies, as compiled in the study. First priorities were formulated on the basis of common needs for fast reactors and, separately, thermal systems. Table 2 shows a summary of the results obtained over the whole set of fast reactors, and Tables 3 and 4 for the two thermal systems (VHTR and PWR). Values are given as uncertainty ranges within selected energy intervals and only the most significant values are shown.
Table 2: SG26 Summary target accuracies for fast reactors [1]

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Quantity</th>
<th>Energy Range</th>
<th>Current Accuracy (%)</th>
<th>Target Accuracy (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-238</td>
<td>(\sigma_{\text{inel}})</td>
<td>6.07 ÷ 0.498 MeV</td>
<td>10 ÷ 20</td>
<td>2 ÷ 3</td>
</tr>
<tr>
<td></td>
<td>(\sigma_{\text{capt}})</td>
<td>24.8 ÷ 2.04 keV</td>
<td>3 ÷ 9</td>
<td>1.5 ÷ 2</td>
</tr>
<tr>
<td>Pu-241</td>
<td>(\sigma_{\text{fiss}})</td>
<td>1.35 MeV ÷ 454 eV</td>
<td>8 ÷ 20</td>
<td>2 ÷ 3 (SFR,GFR,LFR) 5 ÷ 8 (ABTR,EFR)</td>
</tr>
<tr>
<td>Pu-239</td>
<td>(\sigma_{\text{capt}})</td>
<td>498 ÷ 2.04 keV</td>
<td>7 ÷ 15</td>
<td>4 ÷ 7</td>
</tr>
<tr>
<td>Pu-240</td>
<td>(\sigma_{\text{fiss}})</td>
<td>1.35 ÷ 0.498 MeV</td>
<td>6</td>
<td>1.5 ÷ 2</td>
</tr>
<tr>
<td></td>
<td>(\nu)</td>
<td>1.35 ÷ 0.498 MeV</td>
<td>4</td>
<td>1 ÷ 3</td>
</tr>
<tr>
<td>Pu-242</td>
<td>(\sigma_{\text{fiss}})</td>
<td>2.23 ÷ 0.498 MeV</td>
<td>19 ÷ 21</td>
<td>3 ÷ 5</td>
</tr>
<tr>
<td>Pu-238</td>
<td>(\sigma_{\text{fiss}})</td>
<td>1.35 ÷ 0.183 MeV</td>
<td>17</td>
<td>3 ÷ 5</td>
</tr>
<tr>
<td>Am-242m</td>
<td>(\sigma_{\text{fiss}})</td>
<td>1.35 MeV ÷ 67.4 keV</td>
<td>17</td>
<td>3 ÷ 4</td>
</tr>
<tr>
<td>Am-241</td>
<td>(\sigma_{\text{fiss}})</td>
<td>6.07 ÷ 2.23 MeV</td>
<td>12</td>
<td>3</td>
</tr>
<tr>
<td>Cm-244</td>
<td>(\sigma_{\text{fiss}})</td>
<td>1.35 ÷ 0.498 MeV</td>
<td>50</td>
<td>5</td>
</tr>
<tr>
<td>Cm-245</td>
<td>(\sigma_{\text{fiss}})</td>
<td>183 ÷ 67.4 keV</td>
<td>47</td>
<td>7</td>
</tr>
<tr>
<td>Fe-56</td>
<td>(\sigma_{\text{inel}})</td>
<td>2.23 ÷ 0.498 MeV</td>
<td>16 ÷ 25</td>
<td>3 ÷ 6</td>
</tr>
<tr>
<td>Na-23</td>
<td>(\sigma_{\text{inel}})</td>
<td>1.35 ÷ 0.498 MeV</td>
<td>28</td>
<td>4 ÷ 10</td>
</tr>
<tr>
<td>Pb-206</td>
<td>(\sigma_{\text{inel}})</td>
<td>2.23 ÷ 1.35 MeV</td>
<td>14</td>
<td>3</td>
</tr>
<tr>
<td>Pb-207</td>
<td>(\sigma_{\text{inel}})</td>
<td>1.35 ÷ 0.498 MeV</td>
<td>11</td>
<td>3</td>
</tr>
<tr>
<td>Si-28</td>
<td>(\sigma_{\text{inel}})</td>
<td>6.07 ÷ 1.35 MeV</td>
<td>14 ÷ 50</td>
<td>3 ÷ 6</td>
</tr>
<tr>
<td></td>
<td>(\sigma_{\text{capt}})</td>
<td>19.6 ÷ 6.07 MeV</td>
<td>53</td>
<td>6</td>
</tr>
</tbody>
</table>

Table 2 shows several relevant characteristics. Very tight requirements are shown for the \(\sigma_{\text{inel}}\) of \(^{238}\)U (2-3%), Fe-56 (3-6%), Na-23 (4-10%) and even for Pb isotopes. The required accuracies are probably beyond achievable limits with current techniques. These results should be used with caution.
They indicate trends and general priority needs. In fact, these quantitative values have been obtained considering only diagonal (variance) uncertainty values that represent an underestimation of the real uncertainty. Moreover, and certainly more important, the accuracy requirements and priorities are strongly dependent on the assumed initial uncertainty variance-covariance data, in particular in the case of the BOLNA matrix. More recent covariance matrix data are now available, and this type of study should be repeated in order to update the results.

Table 3: SG26 VHTR: Uncertainty reduction requirements needed to meet integral parameter target accuracies

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Quantity</th>
<th>Energy Range</th>
<th>Uncertainty (%)</th>
<th>Initial</th>
<th>Required</th>
</tr>
</thead>
<tbody>
<tr>
<td>U238</td>
<td>$\sigma_{\text{capt}}$</td>
<td>454 - 22.6 eV</td>
<td>1.7</td>
<td>1.2</td>
<td></td>
</tr>
<tr>
<td>C</td>
<td>$\sigma_{\text{inel}}$</td>
<td>19.6 - 6.07 MeV</td>
<td>30.0</td>
<td>7.1</td>
<td></td>
</tr>
<tr>
<td>C</td>
<td>$\sigma_{\text{capt}}$</td>
<td>19.6 - 6.07 MeV</td>
<td>20.0</td>
<td>7.1</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>4 - 0.54 eV</td>
<td>20.0</td>
<td>5.0</td>
<td></td>
</tr>
<tr>
<td>Pu239</td>
<td>$\sigma_{\text{capt}}$</td>
<td>0.54 eV - 0.1 eV</td>
<td>1.4</td>
<td>0.9</td>
<td></td>
</tr>
<tr>
<td>Pu241</td>
<td>$\sigma_{\text{fiss}}$</td>
<td>454 - 22.6 eV</td>
<td>19.4</td>
<td>6.4</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>4 - 0.54 eV</td>
<td>26.8</td>
<td>9.4</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.54 eV - 0.1 eV</td>
<td>2.9</td>
<td>1.5</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.1 eV - thermal</td>
<td>3.3</td>
<td>1.9</td>
<td></td>
</tr>
</tbody>
</table>

Table 4: SG26 PWR: Uncertainty reduction requirements needed to meet integral parameter target accuracies

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Quantity</th>
<th>Energy Range</th>
<th>Uncertainty (%)</th>
<th>Initial</th>
<th>Required</th>
</tr>
</thead>
<tbody>
<tr>
<td>O</td>
<td>$\sigma_{\text{capt}}$</td>
<td>19.6 - 6.07 MeV</td>
<td>100.0</td>
<td>12.1</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>6.07 - 2.23 MeV</td>
<td>100.0</td>
<td>9.9</td>
<td></td>
</tr>
<tr>
<td>Pu241</td>
<td>$\sigma_{\text{fiss}}$</td>
<td>454 - 22.6 eV</td>
<td>19.4</td>
<td>4.7</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>4 - 0.54 eV</td>
<td>26.8</td>
<td>7.7</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.54 eV - 0.1 eV</td>
<td>2.9</td>
<td>1.7</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.1 eV - thermal</td>
<td>3.3</td>
<td>1.9</td>
<td></td>
</tr>
<tr>
<td>Pu239</td>
<td>$\sigma_{\text{capt}}$</td>
<td>0.54 eV - 0.1 eV</td>
<td>1.4</td>
<td>1.0</td>
<td></td>
</tr>
<tr>
<td>U238</td>
<td>$\sigma_{\text{capt}}$</td>
<td>24.8 - 9.12 keV</td>
<td>9.4</td>
<td>4.6</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>454 - 22.6 eV</td>
<td>1.7</td>
<td>1.4</td>
<td></td>
</tr>
<tr>
<td>U238</td>
<td>$\sigma_{\text{inel}}$</td>
<td>6.07 - 2.23 MeV</td>
<td>14.6</td>
<td>5.1</td>
<td></td>
</tr>
<tr>
<td>Pu241</td>
<td>$\sigma_{\text{capt}}$</td>
<td>0.54 eV - 0.1 eV</td>
<td>6.8</td>
<td>3.0</td>
<td></td>
</tr>
<tr>
<td>Pu240</td>
<td>$\sigma_{\text{capt}}$</td>
<td>0.1 eV - thermal</td>
<td>4.8</td>
<td>3.1</td>
<td></td>
</tr>
<tr>
<td>O</td>
<td>$\sigma_{\text{inel}}$</td>
<td>6.07 - 2.23 MeV</td>
<td>54.9</td>
<td>12.6</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>19.6 - 6.07 MeV</td>
<td>84.6</td>
<td>15.6</td>
<td></td>
</tr>
</tbody>
</table>
In fact, more recent studies [3] [4] use a more general method since it accounts for energy correlations in the definition of target accuracies [4] and are based on a recent release of new covariance data: COMMARA 2.0 [5]. The study in [3] was carried out only for minor actinide burners (SFR and ADMAB). In order to have a more compact way to assess the target accuracy, a 7-group energy structure was adopted for this study. The 7-energy group structure is shown in Table 5.

<table>
<thead>
<tr>
<th>Group</th>
<th>Upper Energy (eV)</th>
<th>Group</th>
<th>Upper Energy (eV)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.96403 × 10⁷</td>
<td>5</td>
<td>2.03468 × 10³</td>
</tr>
<tr>
<td>2</td>
<td>2.23130 × 10⁶</td>
<td>6</td>
<td>2.26033 × 10¹</td>
</tr>
<tr>
<td>3</td>
<td>4.97871 × 10⁵</td>
<td>7</td>
<td>5.40000 × 10⁻¹</td>
</tr>
<tr>
<td>4</td>
<td>6.73795 × 10⁴</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

The target accuracies, obtained with the method in [4], on neutron cross sections needed to meet the design requirements (in this particular case ±0.3% \( \Delta k/k \) of uncertainty for the \( k_{eff} \)), are given in Tables 6 and 7. The requirements are shown according (ranked) to their overall original contribution to the total uncertainty on \( k_{eff} \).

The requirements indicated by this study, and in particular for MA, are very small and quite improbable to be attained with current experimental techniques for differential measurements. The results of this investigation suggest that a careful analysis is still needed in order to define the most appropriate and effective strategy for data uncertainty reduction. Besides a further consolidation of the present covariance data libraries, a strategy of combined use of differential and integral measurements should be further pursued in order to meet future requirements, as proposed, e.g. in [6].
Table 6: Nuclear data target accuracy requirements (first 20 major contributors) on a total uncertainty of 300 pcm on $K_{eff}$ for the SFR system

<table>
<thead>
<tr>
<th>Rank</th>
<th>Quantity</th>
<th>Energy Range (eV)</th>
<th>Current (%)</th>
<th>Required (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>$^{240}\text{Pu }\nu$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>3.2</td>
<td>0.9</td>
</tr>
<tr>
<td>2</td>
<td>$^{245}\text{Cm }\sigma_{\text{fiss}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>47.0</td>
<td>3.2</td>
</tr>
<tr>
<td>3</td>
<td>$^{56}\text{Fe }\sigma_{\text{inel}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>11.4</td>
<td>1.8</td>
</tr>
<tr>
<td>4</td>
<td>$^{56}\text{Fe }\sigma_{\text{el}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>8.9</td>
<td>1.7</td>
</tr>
<tr>
<td>5</td>
<td>$^{245}\text{Cm }\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>47.4</td>
<td>4.5</td>
</tr>
<tr>
<td>6</td>
<td>$^{240}\text{Pu }\nu$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>4.8</td>
<td>1.5</td>
</tr>
<tr>
<td>7</td>
<td>$^{240}\text{Pu }\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>2.5</td>
<td>1.1</td>
</tr>
<tr>
<td>8</td>
<td>$^{238}\text{U }\sigma_{\text{inel}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>16.7</td>
<td>2.8</td>
</tr>
<tr>
<td>9</td>
<td>$^{245}\text{Cm }\sigma_{\text{fiss}}$ Gr. 4</td>
<td>$6.74\times10^4$ to $2.03\times10^3$</td>
<td>15.9</td>
<td>3.7</td>
</tr>
<tr>
<td>10</td>
<td>$^{56}\text{Fe }\sigma_{\text{capt}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>12.2</td>
<td>2.7</td>
</tr>
<tr>
<td>11</td>
<td>$^{56}\text{Fe }\sigma_{\text{el}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>5.8</td>
<td>2.1</td>
</tr>
<tr>
<td>12</td>
<td>$^{240}\text{Pu }\nu$ Gr. 1</td>
<td>$1.96\times10^7$ to $2.23\times10^6$</td>
<td>2.5</td>
<td>1.5</td>
</tr>
<tr>
<td>13</td>
<td>$^{238}\text{U }\sigma_{\text{fiss}}$ Gr. 1</td>
<td>$1.96\times10^7$ to $2.23\times10^6$</td>
<td>19.4</td>
<td>3.7</td>
</tr>
<tr>
<td>14</td>
<td>$^{240}\text{Pu }\sigma_{\text{fiss}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>6.2</td>
<td>2.0</td>
</tr>
<tr>
<td>15</td>
<td>$^{240}\text{Pu }\sigma_{\text{capt}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>33.4</td>
<td>4.1</td>
</tr>
<tr>
<td>16</td>
<td>$^{242}\text{Pu }\sigma_{\text{capt}}$ Gr. 4</td>
<td>$6.74\times10^4$ to $2.03\times10^3$</td>
<td>20.2</td>
<td>3.9</td>
</tr>
<tr>
<td>17</td>
<td>$^{244}\text{Cm }\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>14.4</td>
<td>3.2</td>
</tr>
<tr>
<td>18</td>
<td>$^{242}\text{Pu }\sigma_{\text{capt}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>28.2</td>
<td>4.6</td>
</tr>
<tr>
<td>19</td>
<td>$^{240}\text{Pu }\sigma_{\text{capt}}$ Gr. 4</td>
<td>$6.74\times10^4$ to $2.03\times10^3$</td>
<td>4.5</td>
<td>2.1</td>
</tr>
<tr>
<td>20</td>
<td>$^{240}\text{Pu }\nu$ Gr. 4</td>
<td>$6.74\times10^4$ to $2.03\times10^3$</td>
<td>4.6</td>
<td>2.2</td>
</tr>
</tbody>
</table>
Table 7: Nuclear data target accuracy requirements (first 20 major contributors) on a total uncertainty of 300 pcm on $K_{\text{eff}}$ for the ADMAB system

<table>
<thead>
<tr>
<th>Rank</th>
<th>Quantity</th>
<th>Energy Range (eV)</th>
<th>Current (%)</th>
<th>Required (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>$^{243}\text{Cm}$ $\sigma_{\text{fiss}}$ Gr. 3</td>
<td>$4.98\times10^3$ to $6.74\times10^4$</td>
<td>47.0</td>
<td>1.5</td>
</tr>
<tr>
<td>2</td>
<td>$^{244}\text{Cm}$ $\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^4$ to $4.98\times10^5$</td>
<td>14.3</td>
<td>1.0</td>
</tr>
<tr>
<td>3</td>
<td>$^{245}\text{Cm}$ $\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^4$ to $4.98\times10^5$</td>
<td>47.5</td>
<td>2.0</td>
</tr>
<tr>
<td>4</td>
<td>$^{246}\text{Cm}$ $\nu$ Gr. 2</td>
<td>$2.23\times10^4$ to $4.98\times10^5$</td>
<td>8.7</td>
<td>0.9</td>
</tr>
<tr>
<td>5</td>
<td>$^{244}\text{Cm}$ $\sigma_{\text{capt}}$ Gr. 4</td>
<td>$6.74\times10^4$ to $2.03\times10^5$</td>
<td>64.4</td>
<td>2.5</td>
</tr>
<tr>
<td>6</td>
<td>$^{243}\text{Am}$ $\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^4$ to $4.98\times10^5$</td>
<td>8.8</td>
<td>1.0</td>
</tr>
<tr>
<td>7</td>
<td>$^{245}\text{Cm}$ $\sigma_{\text{fiss}}$ Gr. 4</td>
<td>$6.74\times10^4$ to $2.03\times10^5$</td>
<td>14.7</td>
<td>1.6</td>
</tr>
<tr>
<td>8</td>
<td>$^{244}\text{Cm}$ $\sigma_{\text{capt}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>48.6</td>
<td>2.7</td>
</tr>
<tr>
<td>9</td>
<td>$^{244}\text{Cm}$ $\sigma_{\text{fiss}}$ Gr. 1</td>
<td>$1.96\times10^7$ to $2.23\times10^6$</td>
<td>19.7</td>
<td>1.8</td>
</tr>
<tr>
<td>10</td>
<td>$^{243}\text{Am}$ $\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>21.6</td>
<td>2.0</td>
</tr>
<tr>
<td>11</td>
<td>$^{243}\text{Am}$ $\sigma_{\text{capt}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>6.8</td>
<td>1.2</td>
</tr>
<tr>
<td>12</td>
<td>$^{243}\text{Am}$ $\sigma_{\text{fiss}}$ Gr. 1</td>
<td>$1.96\times10^7$ to $2.23\times10^6$</td>
<td>12.1</td>
<td>1.6</td>
</tr>
<tr>
<td>13</td>
<td>$^{243}\text{Am}$ $\sigma_{\text{capt}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>8.2</td>
<td>1.3</td>
</tr>
<tr>
<td>14</td>
<td>$^{244}\text{Cm}$ $\sigma_{\text{capt}}$ Gr. 2</td>
<td>$2.23\times10^4$ to $4.98\times10^5$</td>
<td>77.9</td>
<td>4.3</td>
</tr>
<tr>
<td>15</td>
<td>$^{244}\text{Cm}$ $\nu$ Gr. 1</td>
<td>$1.96\times10^7$ to $2.23\times10^6$</td>
<td>10.1</td>
<td>1.7</td>
</tr>
<tr>
<td>16</td>
<td>$^{244}\text{Cm}$ $\sigma_{\text{fiss}}$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>10.0</td>
<td>2.0</td>
</tr>
<tr>
<td>17</td>
<td>$^{209}\text{Bi}$ $\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^4$ to $4.98\times10^5$</td>
<td>14.0</td>
<td>1.9</td>
</tr>
<tr>
<td>18</td>
<td>$^{15}\text{N}$ $\sigma_{\text{fiss}}$ Gr. 2</td>
<td>$2.23\times10^6$ to $4.98\times10^5$</td>
<td>2.9</td>
<td>0.9</td>
</tr>
<tr>
<td>19</td>
<td>$^{245}\text{Cm}$ $\sigma_{\text{fiss}}$ Gr. 1</td>
<td>$1.96\times10^7$ to $2.23\times10^6$</td>
<td>38.0</td>
<td>4.1</td>
</tr>
<tr>
<td>20</td>
<td>$^{244}\text{Cm}$ $\nu$ Gr. 3</td>
<td>$4.98\times10^5$ to $6.74\times10^4$</td>
<td>10.0</td>
<td>1.9</td>
</tr>
</tbody>
</table>
2.2 Method of JENDL-4.0 Evaluation

In JENDL-4.0 [7], covariances are given for 95 nuclides, i.e. $^{10,11}$B, $^{14,15}$N, $^{16}$O, $^{23}$Na, $^{48}$Ti, $^{52,53}$Cr, $^{55,56}$Fe, $^{58,60}$Ni, $^{90}$Zr, $^{209}$Bi, and all actinides. Except for the actinides, most of the covariances were taken from JENDL-3.3 [8] or additional work [9]. Updates were made to $^{23}$Na and $^{56}$Fe by considering the difference between the evaluated and experimental data. Only the variances of the (n,p) cross-section were provided for $^{14}$N in a limited energy range. Covariances of the Resolved Resonance Parameters (RRPs) of $^{52,53}$Cr and $^{55}$Mn were taken from the resonance analyses [10] [11]. The rest of this section is devoted to the covariance estimation of actinides.

2.2.1 Resonance region

Covariances are given only for Resolved Resonance Parameters (RRPs). Covariances of Unresolved Resonance Parameters (URPs) are not produced, since URP s are used solely for self-shielding calculations. In some cases, the cross-section covariances calculated from the covariances of RRP s are given in order to save file space. For $^{232}$Th, $^{233,235,238}$U and $^{239,241}$Pu, the RRP s evaluated by the ORNL group with the SAMMY code [12] were adopted. For $^{233, 235, 238}$U and $^{239}$Pu, the covariances of RRP s obtained with the SAMMY code [13] were partly adopted. However, after the release of JENDL-4.0 in 2010, it was realised that such truncation of covariance matrices led to incorrect cross-section covariances due to strong long-range correlations of the ORNL results. Therefore, updates were made to the resonance region for those nuclides by considering the full matrices [14]. As a result, the MF32 data were completely replaced with the MF33 data, since it takes a very long time to process such full matrices with the NJOY code [15]. Figure 1 shows the covariance matrix of $^{235}$U fission cross-sections calculated from RRP covariances. For $^{232}$Th and $^{241}$Pu, no covariances were available. Therefore, roughly assumed cross-section variances were given for both nuclides in the resonance region.

Figure 1: Covariance matrix of $^{235}$U fission cross-section

For $^{236,238}$Np, the covariances of RRP s were taken from the SAMMY calculations. For many other isotopes, only variances of the RRP s were given. The variances were taken from the recommendation of Mughabghab [16] and/or from the references on which the parameters were given. If no information was available, uncertainties were assumed tentatively: 0.1% for resonance energies, 10-50% for resonance widths. In some cases, additional cross-section uncertainties were considered to compensate for too small uncertainties calculated from the covariances of RRP s by considering the spread of experimental data or the difference between experimental and evaluated data.
2.2.2 Fission cross-sections above resonance region

For $^{233,235,238}$U and $^{239,240,241}$Pu, the covariance matrices were obtained together with the fission cross-sections using a simultaneous evaluation code SOK [17]. For example, Figure 2 shows the uncertainties obtained for the fission cross section of $^{239}$Pu. Since many experimental data points were considered without correlation among different data sets, the variances obtained were found to be very small. Therefore, the standard deviations were multiplied by a factor of 2, and then adopted in JENDL-4.0. This modification might compensate for the neglected correlation.

Figure 2: Fission cross-sections of $^{239}$Pu (lower panel shows relative difference of cross-sections between JENDL-3.3 and JENDL-4.0)

For other 24 nuclides with many experimental data, the GMA code [18] was used to evaluate the fission cross-sections and their covariances. The standard deviations obtained were compared with experimental data. If the standard deviations were too small, a suitable scaling factor was applied to them. An example is shown in Figure 3.

Figure 3: Fission cross-sections of $^{238}$Pu (the thin solid lines are the standard deviations of JENDL-4.0)
2.2.3 CCONE Calculations

For the data evaluated using the CCONE code [19], covariance matrices were estimated using the KALMAN code [20]. Sensitivities to the model parameters were calculated by the CCONE code using the same parameter sets in the cross-section evaluations. About 50 parameters were used for the KALMAN calculation. The parameters used are those of optical model, level density, fission barrier, γ-ray strength function, and exciton model.

For the capture cross-sections of 232Th, 233-236U, 237Np and 241,243Am, covariance matrices were evaluated using the experimental data sets considered in the cross section evaluations. The statistical and systematic uncertainties of the experimental data were used in the covariance estimations. For the systematic uncertainties, the correlations between the data in the same data set were set to 1.0 and those in the different data sets were assumed to be 0.8 and 0.4 for the same and different authors, respectively. The experimental uncertainties were modified so that the reduced $\chi^2$ is close to unity. The difference between the prior cross-sections, which were calculated with the CCONE code, and the posterior KALMAN estimates was added to the posterior covariances obtained from KALMAN. This was required since the covariance estimation was performed after the JENDL-4.0 cross-sections were fixed. Figure 4 shows the capture cross-section of 237Np. The evaluated uncertainties are shown by the shaded area. The data of JENDL-3.3 and ENDF/B-VII.0 agree with the JENDL-4.0 within the uncertainty in the lower energy region. The agreement among them becomes worse in the upper energy region, where the experimental data are not available.

Figure 4: Capture cross-section of 237Np

For other reaction cross-sections, the covariances were evaluated by a simpler procedure. For the cross-sections for which experimental data were available, the cross-section uncertainties were estimated from the experimental data at selected energy points. Using the estimated uncertainties, the covariance matrices were calculated using the KALMAN code. Figure 5 shows the 238U(n,2n) reaction cross-section with the evaluated uncertainties. The evaluated uncertainties are around 10% except for energies near the threshold. For this evaluation, 10% statistical and systematic uncertainties at 10 and 14 MeV were applied in the KALMAN calculation.
In the upper panel, the evaluated uncertainty is shown by a shaded area. The lower panel shows the relative standard deviation of JENDL-4.0.

For the data for which no experimental data were available, the covariances were obtained using assumed parameter uncertainties. The parameter uncertainties were estimated from available experimental data on neighbouring nuclides. In most cases, however, the uncertainties thus obtained were increased by a factor of two when they were applied to a reaction for which experimental data were unavailable. A typical uncertainty was several percent depending on the sensitivity of the cross-section to the parameter.

2.2.4 Mean prompt- and delayed-neutron multiplicity

For the nuclides having enough experimental data, the $\nu_p$ value and its covariance were determined by the least-squares fitting to the experimental data. In some cases, a suitable scaling factor was applied to the variances obtained when they were too small. If no experimental data were available, the systematics recommended by Howerton [21] or Ohsawa [22] were applied to estimate $\nu_p$ values. In such cases, the covariances were estimated by fitting a linear function to two data points at 0 MeV and 5 MeV calculated from the systematics. Their uncertainties were assumed to be 5% for nuclides lighter than Cm and 10% for the nuclides heavier than Bk. If possible, the uncertainties were determined from experimental data at the thermal energy.

For the $\nu_d$ values, only the variances were given except for $^{235,238}$U and $^{239}$Pu for which the JENDL-3.3 covariances obtained by the GMA analyses were adopted. Variances were estimated from experimental data when available. A standard deviation of 15% was assumed if no experimental data were available at all.
3. Review of experiments

Recent progress on cross-section measurements for neutron induced reactions is reviewed. Due to the use of high intensity spallation neutron beams, improved detection systems and elaborated analysis techniques, a substantial improvement has been made in the experimental techniques. The status on capture cross-section measurements is reviewed in Section 3.1. In this section also, a review on experimental data that is available to evaluate the capture cross-section for $^{28}$Si, $^{206}$Pb, $^{238}$U, $^{241}$Am, $^{243}$Am and $^{237}$Np is given. The status of the $^{10}$B(n,α) reaction cross-section is discussed in Section 3.2. The status of inelastic cross section measurements is briefly reviewed in Section 3.3. New experimental approaches using inverse kinematics methods are described in Section 3.4 because of their potential impact especially on fission data for unstable nuclei.

3.1 Current status of capture cross-section measurements

Neutron induced capture cross-section measurements rely either on post-irradiation activation analysis or on the detection of prompt $\gamma$-rays emitted in the (n,$\gamma$) reaction. To cover a broad energy region, measurements can be performed at a white neutron source applying the time-of-flight (TOF) technique or alternatively by using quasi-mono-energetic neutrons produced by charged particle induced reactions. The choice of the principle and related detection system depends on the reaction to be studied, the energy region of interest, the amount of available sample material and the required accuracy and resolution. Capture cross-sections in the resonance region are best derived from results of experiments with a prompt $\gamma$-ray detection system that is optimised for TOF measurements [23]. Post-irradiation activation analysis is suited to determine capture cross-sections at thermal energies and in the continuum region, and to derive experimental resonance integrals. Recently, the activation method has also been applied to determine neutron cross-sections in a fast reactor spectrum.

3.1.1 Prompt $\gamma$-ray detection method

A prompt $\gamma$-ray detection system optimised for TOF-measurements fulfils the following requirements [23]:

- the detection efficiency for a capture event is independent of the $\gamma$-ray cascade, i.e. independent of the multiplicity of the $\gamma$-ray spectrum and the $\gamma$-ray energy distribution;
- the sensitivity to neutrons scattered by the sample is low compared to the sensitivity to $\gamma$-rays produced by the capture reaction in the sample;
- the detector has a good time resolution;
- for the study of a fissioning nucleus, the $\gamma$-rays from neutron capture can be separated from those resulting from fission;
- in the case of a radioactive sample, the prompt $\gamma$-rays can be separated from the $\gamma$-rays emitted due to the radioactive decay.
Three different principles based on the direct detection of prompt \( \gamma \)-rays can be distinguished \cite{23}: (1) \( \gamma \)-ray spectroscopy (GS), (2) total \( \gamma \)-ray absorption (TA) and (3) total energy detection principle (TE). The main uncertainty for the three principles is related to the normalisation of the data and the determination of the background \cite{23}.

(1) Capture cross-sections based on \( \gamma \)-ray spectroscopic measurements with high resolution \( \gamma \)-ray detectors \cite{23} \cite{25} can be derived from:

- the sum of all the partial cross-sections of primary transitions depopulating the capture state (GS1);
- the sum of the partial capture cross-sections of the transitions feeding the ground state (GS2); or
- the sum of all the observed partial cross-sections weighted with the energy of the transition divided by the total \( \gamma \)-ray energy liberated in the capture event (GS3).

The accuracy strongly depends on the complexity of the level scheme of the compound nucleus. The cross-section can be determined accurately when the \( \gamma \)-ray transitions of the cascade are well known. Therefore, \( \gamma \)-ray spectroscopic methods are very powerful to determine capture cross-section data for light nuclei or for nuclei with a proton and neutron number close to a magic shell \cite{24} \cite{25}. When not all \( \gamma \)-ray transitions can be determined, the results are biased and only lower limits can be derived \cite{26}. To verify the impact of missing transitions, the principle of \( \gamma \)-ray intensity balance \cite{27} or crossing intensity sum \cite{28} can be applied. The missing contributions can also be based on statistical models to simulate the full \( \gamma \)-ray cascade. Codes that can be used are e.g. DICEBOX \cite{29}, DECAYGEN \cite{30} and \( \gamma \)DEX \cite{31}. The \( \gamma \)-ray cascade simulations rely on nuclear level statistical models and nuclear data input (low-lying level scheme, average radiation widths and level densities). Using spectroscopic measurements, the accuracy of the cross-section depends on the statistical nature of the \( \gamma \)-ray cascade.

(2) The total \( \gamma \)-ray absorption principle relies on the detection of the energy sum of the \( \gamma \)-rays emitted in a capture event. An ideal detector has a 4\( \pi \) geometry and a 100\% absolute detection efficiency allowing for the detection of the entire electromagnetic cascade. Thus, the energy deposited in the detector is directly proportional to the total energy available in the capture event and independent of the \( \gamma \)-ray cascade.

The first total absorption detectors were large liquid organic scintillation tanks \cite{32} \cite{33}. The uncertainty of these systems is limited to 5-10\% and depends on the reaction under study. The limitation is primarily due to corrections that are required to estimate the efficiency to detect a capture event \cite{23} \cite{33}. Organic liquid scintillators (OLS) have extensively been used to determine capture cross-sections of fissile material. To separate capture events from fission events, different methods have been applied. Some of them use an additional fission chamber in parallel or as an additional measurement to determine correction factors. An extensive list of capture-to-fission ratio measurements for \(^{233}\text{U},^{235}\text{U}\) and \(^{239}\text{Pu}\) is given in \cite{23}.

Nowadays, in-organic detectors are used. An overview of systems that are in use is given in \cite{23}. In-organic scintillators are smaller in size and have a better detection efficiency compared to OLS. Therefore their sensitivity to the ambient background is reduced. However, they still suffer from neutron sensitivity due to \((n,\gamma)\) reactions in the detection material. Therefore, they are limited to measurements in the resolved resonance region and for nuclei with small scattering to capture ratios. The final accuracy of such systems depends strongly on the reaction under study. Since an ideal detector with a 100\% \( \gamma \)-ray detection efficiency does not exist, a correction is needed when the normalisation is performed using a capture reaction which has a different \( \gamma \)-ray cascade from the reaction under study. Such a correction becomes even more important when a constraint is imposed on the multiplicity and energy deposition to reduce the
background and when the $\gamma$-ray cascade changes from resonance to resonance. Due to an improved understanding of the measurement equipment and techniques through Monte Carlo simulations, the detection efficiencies can be determined with better accuracies. However, the final accuracy depends strongly on the statistical nature of the $\gamma$-ray cascade, as in the case of organic scintillators. These detectors can also be used to derive capture-to-fission ratio for fissile materials [34] [35].

(3) When the contribution of the fission channel can be neglected, the most accurate capture cross-section data can be measured by applying the total energy detection principle (TE) using C$_6$D$_6$ detectors combined with the pulse height weighting technique (PHWT). The application of the total energy detection principle requires a $\gamma$-ray detector with a relatively low $\gamma$-ray detection efficiency which is proportional to the $\gamma$-ray energy. Under these conditions the efficiency to detect a capture event is directly proportional to the sum of the energies of the $\gamma$-rays emitted in the cascade. This makes the efficiency in first approximation independent of the $\gamma$-ray cascade.

The Moxon-Rae detector achieves approximately the proportionality between the $\gamma$-ray energy and detection efficiency by a special design of the detector [36]. However, uncertainties due to imperfect linearity between the detection efficiency and the $\gamma$-ray energy are at least 5% [37] [38].

Correction factors in the case of the total energy detection principle combined with PHWT are limited compared with all the other principles (GS or TA). This has a strong impact on the accuracy that can be reached. An experimental validation of the total energy detection principle combined with the PHWT for C$_6$F$_6$ detectors was performed by Yamamuro et al. [39]. Normalisation factors derived from the saturated resonances at 4.3 eV in $^{181}$Ta, 4.9 eV in $^{197}$Au and 5.2 eV in $^{109}$Ag, were consistent within 2% [39]. A more extensive performance assessment for a C$_6$D$_6$ based system has been carried at the GELINA facility of the EC-JRC-IRMM [23] [40] [41]. The results in [23] [40] demonstrate that capture yields with uncertainties better than 2% can be deduced from thermal energy up the URN when the total energy detection principle in combination with the PHWT is applied. However, such a low uncertainty can only be reached under specific constraints, as discussed in [23].

3.1.2 Post-irradiation activation method

Activation method has been widely used to determine thermal neutron capture cross-sections using thermal reactor neutrons. The energy dependent capture cross-sections measured by neutron time-of-flight technique have been sometimes normalised at the thermal cross-sections determined by activation method. Therefore, this technique has an important role in determining neutron capture cross-sections and their uncertainties. Recently, activation method using fast reactor neutrons was applied to deduce fast neutron capture cross-sections in the case of $^{237}$Np [42]. Activation method using accelerator based quasi mono-energetic neutrons was also widely used to determine 25-keV Maxwellian-averaged cross-sections and 14 MeV cross-sections. Activation method is in principle a measurement method independent of prompt $\gamma$-ray detection method. Therefore, cross-check of data measured by these two independent methods is valuable to find systematic errors of unknown origin.

For measurements of thermal neutron capture cross-sections and resonance integrals, the well-known formulation, called as Westcott formulation, has been used when well moderated reactor neutrons are used for activation. In this case, neutron flux distribution is approximated by a Maxwell distribution plus 1/E (inverse of neutron energy) components. The effective capture cross-section $\sigma$ is expressed in this case as:

$$\sigma = \sigma_0 (g + r s),$$

where $\sigma_0$ is the cross-section for 2 200 m/sec neutrons and g and s are functions of the temperature T depending on the departure of the cross-section law from the 1/v form (for
a $1/v$ law, $g=1$ and $s=0$). The $r$ is an epithermal index that represents the relative strength of the epithermal $1/E$ component. The $s$ is a quantity proportional to the reduced resonance integrals, i.e. resonance integral minus $1/v$ component. By irradiating a sample in two different neutron fields, for example, irradiating it with and without a Cd cover sheet, both quantities $\sigma_0$ and resonance integrals are deduced.

The $g$ is known as the Westcott factor, which represents the departure of the cross-section law from the $1/v$ form. To deduce $g$, energy dependence of the capture cross-section measured by a TOF method is required. Recent TOF measurements with thin samples are expected to give an accurate $g$-factor value, since the correction factors for neutron shields and multiple scatterings are small. Other systematic effects require appropriate corrections, such as dead-time and background corrections, and the corresponding uncertainties need to be propagated [43]. When the thermal capture cross-section is used for normalisation of TOF data, a simultaneous analysis of both kinds of data is desired.

In the case of an activation method, high neutron flux of reactor can be used for irradiation. Using this advantage, double capture reactions [44] and even triple capture reactions [45] have been utilised to deduce capture cross-sections.

In the case that neutron capture product nucleus emits gamma rays, Ge detectors have been currently used to determine the amount of product nucleus. The reliability of gamma-ray emission probability is a possible origin of the unrecognised uncertainty, and should be carefully checked. The situation of reliability on emission probability is similar to the case of neutron capture cross-section [46]. Therefore, precise measurement techniques [47] of gamma-ray emission probability are also important to obtain accurate capture cross-sections.

In the case that neutron capture product nucleus does not emit gamma rays, mass spectrometry [48] has been used to determine the amount of product nucleus. The method has also been used to characterise the sample in the measurement of the thermal neutron capture cross-section of radioactive $^{135}$Cs [49].

The chemical purity of the sample is the other possible origin of the unrecognised uncertainty. For example, an impurity of $^{238}$U was rigorously investigated in the measurement of neutron capture cross-section of $^{238}$Np using double neutron capture reaction [44] using a $^{237}$Np sample, since the effect of contamination of $^{238}$U in the sample is significant; $^{238}$U is transmuted to $^{239}$Np via one neutron capture reaction and a $\beta$ decay process.

Post-irradiation activation analysis has also been performed at quasi-mono energetic neutron beams produced by charged particle induced reactions, where the neutron energy can be controlled by changing the energy of the interacting charged particle and/or the emerging angle. The cross-section can be derived from a measurement of the residual activity by $\gamma$-ray spectroscopy or by accelerator mass spectrometry (AMS) [50].

### 3.1.3 Experimentalist’s review on cross-section data for the capture reaction of $^{28}$Si, $^{206}$Pb, $^{238}$U, $^{241}$Am, $^{243}$Am and $^{237}$Np

In this section, documented cross-section data that can be used to evaluate the capture cross-sections for $^{28}$Si, $^{206}$Pb, $^{238}$U, $^{241}$Am, $^{243}$Am and $^{237}$Np from thermal up to the Unresolved Resonance Region (URR) are discussed. In the Resolved Resonance Region (RRR), reliable resonance parameters can only be derived when transmission data are available [23] [51]. Total cross-section data provide also important prior information to improve the accuracy of capture cross-sections in the URR as shown by Sirakov et al. [52]. Hence, transmission data are needed to perform a consistent evaluation of the capture cross section in the resonance region with uncertainties that are requested in the conclusions of SG26. Therefore, in the discussion of available experimental data results of transmission measurements have also been considered.
28Si

Experimental data that can be used for an evaluation of the thermal capture cross-section of 28Si are summarised in Table 8. The reference cross-section that was used is also given. The value recommended by Raman et al. [25] is fully consistent with the cross-section derived from the three γ-spectroscopic methods mentioned in the introduction of Section 3.1.1. This cross-section is relative to the (332.6 ± 0.6) mb capture cross-section for 1H determined by Cokinos and Melkonian [53]. The capture cross-section reported by Islam et al. [54] deviates by more than 20%. This is partly due to the reference value for 14N(n,γ) used in [54], which is 15% higher compared to the value 68.77 (± 0.56) mb reported by Belgya [28]. For a full consistent evaluation, the coherent scattering length \( b_c = 4.106 (± 0.006) \) fm recommended by Koester et al. [55] can also be used.

| Pomerance [56] | 81 (± 24) mb | 197Au(n,γ) | 95 b |
| Spits and De Boer [57] | 156 (± 23) mb | Al(n,γ) | 239 (± 3.0) mb |
| Spits and De Boer [57] | 163 (± 57) mb | Na(n,γ) | 534 (± 5.0) mb |
| Spits and De Boer [57] | 166 (± 33) mb | Mn(n,γ) | 13.3 (± 0.2) b |
| Islam et al. [54] | 207 (± 4) mb | 14N(n,γ) | 79.8 (± 1.4) mb |
| Raman et al. [25] | 169 (± 4) mb | 1H(n,γ) | 332.6 (± 0.6) mb |
| GS1 | 169.4 (± 3.9) mb | |
| GS2 | 168.0 (± 3.7) mb | |
| GS3 | 168.1 (± 3.9) mb | |

Capture and transmission measurements for 28Si+n have been carried out at GELINA at a 130-m and 400-m flight path, respectively [58]. For the capture measurements, a γ-spectroscopic detection system based on BGO-detectors was used. The capture data were normalised to the 1.15 keV resonance of 56Fe. However, no further details about the normalisation procedure, e.g. the partial radiation width used for the normalisation, were specified. Partial capture cross sections for transitions to the ground state and the first and third excited state have been determined. Resonance areas for 28Si are given for resonance energies up to 4638 keV. Unfortunately the experimental transmission and yields are not available in numerical data. Documented TOF-data resulting from total and capture measurements at ORELA are available in numerical form [59-63]. Transmission measurements on natural samples with different thickness have been carried out at a 47-m, 80-m and 200-m flight path [59-62]. Results of capture measurements at a 40-m station have been reported by Guber et al. [63]. The total energy detection principle in combination with the PHWT using C6D6 detectors was applied. From these data, capture areas and peak cross-sections for neutron energies < 700 keV can be deduced with an accuracy < 5%. However, this level of accuracy cannot be reached for the capture cross-section between resonances. This is due to the contribution of direct capture and/or possible interference effects.

The ORELA data together with the transmission data of Adib et al. [64] have been used in a resonance shape analysis by Derrien et al. [65]. In the analysis, the impact of a direct capture contribution was also considered. The analysis was based on a Reich-Moore approximation of the R-matrix theory. Hence, interference effects for the capture channel have been neglected. The resulting thermal capture cross-section is consistent with the one recommended by Raman et al. [25]. However, the coherent scattering length is about
5% smaller compared to the one recommended by Koester et al. [55]. Although the above mentioned data on $^{28}$Si and the evaluation of Derrien et al. [65] do not cover the energy region of interest highlighted by SG26 in Table 2 (i.e. 6 MeV – 20 MeV), they provide essential data to produce consistent capture cross-sections in the high energy region.

$^{206}$Pb

Experimental capture cross-sections at thermal energy that have been reported in the literature are: $\sigma(\text{n}_{\text{th}},\gamma) = 25.5 \pm 5.0$ mb [66], $30.5 \pm 0.7$ mb [67], $26.6 \pm 1.2$ mb [68] and $29.2 \pm 0.7$ mb [24]. These values together with the coherent scattering lengths $b_c = 9.22 \pm 0.07$ fm determined by Ioffe et al. [69] and $b_c = 9.23 \pm 0.05$ by Koester and Knopf [70] can be used to derive consistent capture and scattering cross-sections at a thermal energy.

Transmission measurements for neutron energies between 1 keV and 900 keV have been carried out at a 78-m and 200-m station of ORELA using radiogenic lead samples enriched to 88.4% in $^{206}$Pb [71-73]. Capture cross-section measurements using Ge-detectors at a 40-m station of ORELA are reported in [71]. Capture and transmission measurements on a pure $^{206}$Pb sample at a 60-m and 25-m station, respectively, have been performed at GELINA by Borella et al. [74]. Capture yields were obtained by applying the total energy detection principle combined with the PHWT using C$_6$D$_6$ detectors. The neutron sensitivity of the set-up was determined by Monte Carlo simulations and verified by experiment. The measurements, data reduction and analysis procedures were carried out following the recommendations in [23]. Results of a simultaneous analysis of capture and transmission data for neutron energies < 80 keV are given in [74]. For neutron energies between 80 keV and 625 keV the experimental yield was analysed by fixing the neutron widths reported by Horen et al. [72] [73]. From these data, capture areas with an accuracy < 5% have been deduced. The thermal data in [24,66-70] and the TOF-data in [71-74], which are available in numerical form in the EXFOR library, can be used to improve the total and capture cross-section for neutron induced reactions in $^{206}$Pb below 650 keV. From such an evaluation, capture areas and peak cross-sections for neutron energies < 200 keV can be deduced with an accuracy < 5%. However, this level of accuracy cannot be reached for the cross-section between resonances. Since Borella et al. [74] and Mizumoto et al. [71] have demonstrated that the $\gamma$-ray emission spectra are limited to a few cascades, the contribution of direct capture and/or interference effects cannot be excluded. Capture measurements using C$_6$D$_6$ detectors on pure $^{206}$Pb have also been reported in [75]. However, the experimental yield is not available in numerical form. In addition, a weighting for an ideal detection system was applied, i.e. for a 0 eV discrimination level on the observed energy deposited in the C$_6$D$_6$ detector. Therefore, correction factors to account for the finite discrimination level are required, which are very difficult to determine accurately for neutron capture on $^{206}$Pb [23].

Capture measurements on $^{238}$U($n,\gamma$) in the energy region below 200 keV

The status of the thermal capture cross-section has been reviewed by Trkov et al. [76]. A list of measured coherent scattering lengths is given in [77]. These scattering lengths together with results of transmission and capture measurements at ORELA have been used by Derrien et al. [78] to determine parameters of individual resonances for $^{238}$U below 20 keV. The transmission experiments were carried out at a 40-m, 150-m and 200-m station and the capture measurements at a 40-m and 150-m station using an OLS. The capture data of Corvi et al. [79] obtained at GELINA together with the thermal capture cross section of Poenitz et al. [80] have been used to adjust the parameters of the bound state(s). The resulting capture cross-section at thermal $\sigma(n_{\text{th}},\gamma) = 2.7$ b is close to the value $\sigma(n_{\text{th}},\gamma) = 2.683 \pm 0.012$ b recommended by Trkov et al. [76].

A list of capture cross-section data (including absolute and shape data) for $^{238}$U that can be used to determine the average capture cross-section in the URR and at a higher energy level is given in [81]. These data have been used to evaluate the average capture cross-section up to 2.2 MeV based on a least-square adjustment using the GMA code.
developed by Poenitz [82]. Average capture cross-sections below 200 keV are recommended with uncertainties between 0.5% and 3.3% in [81]. Other evaluations of the capture cross-section for $^{238}$U below 200 keV are reported by Fröhner [83], Maslov et al. [84] and Courcelle et al. [85]. These evaluations result from a parameterisation of the cross-section data by the Hauser-Feshbach formalism including width fluctuation corrections. The evaluation process includes results of total cross-section and inelastic cross-section data.

Unfortunately only one set of capture cross-section data in the resonance region, the one reported by Yamamuro et al. [86], was based on the total energy detection principle using C$_6$D$_6$ detectors. In addition, this data set suffers from a rather large 7.7% normalisation uncertainty. Recently, capture cross section experiments in the resonance region have been carried out at GELINA and n_TOF as part of the ANDES project [87]. Measurements using C$_6$D$_6$ detectors combined with the PHWT have been carried out at GELINA [88] and n_TOF [89] [90]. A re-evaluation of the cross-section data, including the results of these measurements, should result in a capture cross-section for $^{238}$U with uncertainties around 2% for neutron energies below 200 keV.

$^{241}$Am($n,\gamma$)

A list of experimental data that can be used to re-evaluate the capture cross-section of $^{241}$Am is given in [91] [92]. The data include thermal capture cross-sections, integral measurements and transmission and capture TOF-data. Transmission measurements in the resonance region have been performed by Adamchuk et al. [93], Slaughter et al. [94], Derrien and Lucas [95], Belanova et al. [96], Kalebin et al. [97] and Lampoudis et al. [92]. Results of capture experiments have been reported by Weston and Todd [98], Gayther and Thomas [99], Wisshak and Käppeler [100], Vanpraet and Cornelis [101], Jandel et al. [102] and Lampoudis et al. [92].

Table 9 shows the experimental cross-section data for the capture reaction at 0.0253 eV. Data reported in the literature for the total capture cross-section ($\sigma_{m+g}$), for the capture cross-section to the $1^-$ ground state in $^{242}$Am ($\sigma_{g}$) and for the reaction to the $5^-$ isomeric state at 49 keV of $^{242}$Am ($\sigma_{m}$) are given. The second column specifies the measurement technique that was applied, i.e. neutron activation analysis (A), pile oscillator measurements (P) or time-of-flight experiments (T). The most recent thermal capture cross-section derived by Lampoudis et al. [92] ($\sigma (n_{th},\gamma) = 749 (\pm 35) b$) is 19% and 12% larger than the one derived from the TOF-measurements of Kalebin et al. [97] and Jandel et al. [102], respectively. This systematic discrepancy is reflected in differences with recommended values in the evaluated data libraries, which are also listed in Table 9. The value of Lampoudis et al. [92] is 10% larger than the latest evaluated value (JENDL-4.0), but is within the quoted uncertainties in agreement with results of measurements at ILL [103] [104] and at KURRI [105]. As noted in [106], the cross-section derived from the data in [105] is even underestimated due to the Westcott factor $g = 1.05$ that was used. The overestimation of the Westcott factor $g = 1.05$ recommended by Mughabghab [16] is also confirmed by results from measurements at J-PARC [107].

Figure 6 reveals that the resonance strength of the resonances at 0.306 eV, 0.574 eV and 1.272 eV derived by Derrien and Lucas [95], Kalebin et al. [97], Weston and Todd [98] and Jandel et al. [102] are systematically lower compared to those of Lampoudis et al. [92].
Table 9: Experimental data for the thermal capture cross-section of $^{241}$Am

<table>
<thead>
<tr>
<th>References</th>
<th>Method</th>
<th>$\sigma_m+g$ (b)</th>
<th>$\sigma_g$ (b)</th>
<th>$\sigma_m$ (b)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pomerance [108]</td>
<td>P</td>
<td>628.5 ± 35</td>
<td>670 ± 6</td>
<td>70 ± 5</td>
</tr>
<tr>
<td>Bak et al. [109]</td>
<td>A</td>
<td>740 ± 60</td>
<td>573 ± 103</td>
<td>74 ± 15</td>
</tr>
<tr>
<td>Dovbenko et al. [110]</td>
<td>A</td>
<td>624 ± 20</td>
<td>670 ± 60</td>
<td>70 ± 5</td>
</tr>
<tr>
<td>Harbour et al. [111]</td>
<td>T</td>
<td>612 ± 25</td>
<td>670 ± 60</td>
<td>70 ± 5</td>
</tr>
<tr>
<td>Kalebin et al. [97]</td>
<td>A</td>
<td>624 ± 20</td>
<td>670 ± 60</td>
<td>70 ± 5</td>
</tr>
<tr>
<td>Gavrilov et al. [112]</td>
<td>A</td>
<td>673 ± 10</td>
<td>602 ± 9</td>
<td>60 ± 4</td>
</tr>
<tr>
<td>Shinohara et al. [113]</td>
<td>A</td>
<td>696 ± 46</td>
<td>636 ± 46</td>
<td>60 ± 4</td>
</tr>
<tr>
<td>Maidana et al. [114]</td>
<td>A</td>
<td>696 ± 46</td>
<td>636 ± 46</td>
<td>60 ± 4</td>
</tr>
<tr>
<td>Fioni et al. [103]</td>
<td>A</td>
<td>705 ± 23</td>
<td>628 ± 22</td>
<td>60 ± 4</td>
</tr>
<tr>
<td>Nakamura et al. [105]</td>
<td>A</td>
<td>702 ± 25</td>
<td>628 ± 22</td>
<td>60 ± 4</td>
</tr>
<tr>
<td>Jandel et al. [102]</td>
<td>T</td>
<td>665 ± 33</td>
<td>540 ± 32</td>
<td>60 ± 4</td>
</tr>
<tr>
<td>ENDL/B-VI.8</td>
<td></td>
<td>620 ± 13</td>
<td>647 ± 34</td>
<td>64.8 ± 7.8</td>
</tr>
<tr>
<td>JEFF-3.1.2</td>
<td></td>
<td>639.5</td>
<td>647 ± 34</td>
<td>64.8 ± 7.8</td>
</tr>
<tr>
<td>JENDL-3.3</td>
<td></td>
<td>684 ± 15</td>
<td>620.1 ± 7.8</td>
<td>64.8 ± 7.8</td>
</tr>
<tr>
<td>JENDL-4.0</td>
<td></td>
<td>684 ± 15</td>
<td>620.1 ± 7.8</td>
<td>64.8 ± 7.8</td>
</tr>
</tbody>
</table>

The symbols ($\sigma_m+g$, $\sigma_g$, $\sigma_m$) are explained in the text. The experimental method that was used is indicated by P (pile oscillation), A (activation) and T (time-of-flight).

Figure 6: Ratio of resonance strengths ($g^{\Gamma_n}$) derived by Derrien and Lucas [95], Kalebin et al. [97], Weston and Todd [98] and Jandel et al. [102] and the one measured at GELINA by Lampoudis et al. [92]

![Graph showing the ratio of resonance strengths](image)

On the other hand, the resonance integrals calculated with the parameters of Lampoudis et al. [92] are fully consistent with the results of direct measurements [103] [106] [110] [112-114] as illustrated in Figure 7. This suggests that the resonance strengths reported by Derrien and
Lucas [95], Kalebin et al. [97], Weston and Todd [98] and Jandel et al. [102] are underestimated. Such an underestimation might occur when powder samples are used and no correction for the particle size is applied. As discussed in [23], an underestimation of the neutron width when using powder samples will coincide with an overestimation of the radiation width.

Figure 7: The resonance integral for the $^{241}$Am(n,γ) as a function of the cut-off energy

A comparison of the data in Table 10 confirms that Derrien and Lucas [95] and Weston and Todd [98] deduced larger radiation widths from their data. However, this does not explain the smaller resonance strengths from Kalebin et al. [97] and Jandel et al. [102]. Lampoudis et al. [92] suggest that the resonance strengths and thermal value of Jandel et al. [102] are underestimated due the normalisation procedure that is applied.

**Table 10: Comparison of the average radiation width for $^{241}$Am + n reported by Derrien and Lucas [95], Kalebin et al. [97], Weston and Todd [98], Jandel et al. [102] and Lampoudis et al. [92]**

<table>
<thead>
<tr>
<th>References</th>
<th>Average radiation width</th>
</tr>
</thead>
<tbody>
<tr>
<td>Derrien and Lucas</td>
<td>44.2 ± 0.1 meV</td>
</tr>
<tr>
<td>Kalebin et al.</td>
<td>42.9 ± 0.3 meV</td>
</tr>
<tr>
<td>Weston and Todd</td>
<td>47.6 ± 0.2 meV</td>
</tr>
<tr>
<td>Jandel et al.</td>
<td>43.8 ± 1.3 meV</td>
</tr>
<tr>
<td>Lampoudis et al.</td>
<td>42.1 ± 0.3 meV</td>
</tr>
</tbody>
</table>

The accuracy of both the resonance parameters and thermal capture cross-section can be improved by re-analysing the above mentioned TOF-data together with results of capture measurements at J-PARC [107] and n_TOF [115] and an activation experiment at FRM-II in Garching [116]. Although such a re-evaluation will not directly contribute to an improved capture cross-section in the keV region these data, in particular the transmission data of GELINA, can be used to correct for systematic effects due to sample characteristics and improve the normalisation of the capture data of Weston and Todd [98], Gayther and Thomas [99], Wisshak and Käppeler [100], Vanpraet and Cornelis [101] and Jandel et al. [102].
The experimental data available for the evaluation of the \(^{243}\text{Am}(n,\gamma)\) capture cross-section at thermal energies are summarised in Table 11. As it can be observed, the data are scattered mainly between 75 and 85 barns, and are in some cases incompatible. There is no clear choice about which thermal cross-section \(\sigma_0\) should be used from the point of view of the data. There are also large discrepancies between the different resonance integrals \(I_0 = \int \sigma(E) dE\) obtained from the data.

Table 11: Thermal capture cross-sections, resonance integrals and ratios between them provided by different experiments and evaluations of \(^{243}\text{Am}\)

<table>
<thead>
<tr>
<th>References</th>
<th>(\sigma_0) (barn)</th>
<th>(I_0) (barn)</th>
<th>(I_0/\sigma_0)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hori et al. (2009)</td>
<td>76.6(^{a})</td>
<td>1970(±110)</td>
<td>25.7(±15)</td>
</tr>
<tr>
<td>Marie et al. (2006)</td>
<td>81.8(±36)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ohta et al. (2006)</td>
<td>84.4</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hatakawa et al. (1997)</td>
<td>83(±6)</td>
<td>2200(±150)</td>
<td>26.5(±26)</td>
</tr>
<tr>
<td>Simpson et al. (1974)</td>
<td>77(±2)</td>
<td>1930(±50)</td>
<td>25.1(±9)</td>
</tr>
<tr>
<td>Eberle et al. (1971)</td>
<td>85(±4)</td>
<td>1824(±80)</td>
<td>21.5(±14)</td>
</tr>
<tr>
<td>Folger et al. (1968)</td>
<td>78</td>
<td>2250(±10)</td>
<td>29</td>
</tr>
<tr>
<td>Bak et al. (1967)</td>
<td>73(±6)</td>
<td>2300(±200)</td>
<td>32(±4)</td>
</tr>
<tr>
<td>Ice et al. (1966)</td>
<td>66&lt;(\sigma_0&lt;84)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Butler et al. (1957)</td>
<td>73.6(±1.8)</td>
<td>2290(±50)</td>
<td>31(±1)</td>
</tr>
<tr>
<td>Harvey et al. (1954)</td>
<td>140(±50)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Stevens et al. (1954)</td>
<td>115</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Mughabghab (2006)</td>
<td>75.1(±18)</td>
<td>1820(±70)</td>
<td>24.2(±11)</td>
</tr>
<tr>
<td>ENDF/B-VII.1</td>
<td>80.4</td>
<td>2051</td>
<td>25.5</td>
</tr>
<tr>
<td>ENDF/B-VII.0</td>
<td>75.1</td>
<td>1820</td>
<td>24.2</td>
</tr>
<tr>
<td>JENDL-4.0</td>
<td>79.3</td>
<td>2040</td>
<td>25.7</td>
</tr>
<tr>
<td>JEFF-3.1</td>
<td>76.7</td>
<td>1788</td>
<td>23.3</td>
</tr>
</tbody>
</table>

(a) Value assumed for normalisation. \(I_0\) is proportional to it.
(b) The thermal value of Marie et al. was assumed. The Ohta et al. measured value was \(\sigma_0 = 174.5(±5.3)\) barn and \(\sigma = 0.0418(±45)\), where \(\sigma_0 = \sigma + (0.45 - 1/\sigma)\).
(c) Cut-off energy was taken as 0.625 eV instead of 0.5 eV.
(d) Cut-off energy was taken as 0.83 eV instead of 0.5 eV.

The available capture and transmission differential data are given in Table 12. The information is presented in two groups: measurements before 1985 and recent measurements after 2000, which have not yet been used in any of the evaluations available in 2013. Indeed, the data of Mendoza et al., Jandel et al. and Hori et al. are not yet available in EXFOR (the Mendoza et al. data were accessible for the preparation of this report).
The resolved resonance region extends up to 250 eV in all the most recent evaluations available in 2013 (ENDF/B-VII.1, JEFF-3.1.2, JENDL-4.0, CENDL-3.1, ROSFOND-2010, BROND-2.2). Excluding the fission widths and the strongest resonance at 1.35 eV, the JEFF-3.1.2, JEFF-3.1, JENDL-4.0, JENDL-3.3, BROND-2.2 and CENDL-3.1 libraries are based on the evaluation performed by Maslov et al. [131], and the ENDF/B-VII.1 and ENDF/B-VII.0 evaluations on the values provided by Mughabghab [16] and Simpson et al. [122], respectively. However, all of them are based essentially on the transmission experiment performed by Simpson et al. quite compatible with the Berreth et al. data, but covering a larger energy range.

### Table 12: Differential transmission and capture measurements of $^{243}\text{Am}$

<table>
<thead>
<tr>
<th>Reference</th>
<th>Type of Measurement</th>
<th>Energy Range</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bellanova et al. (1976)</td>
<td>Transmission</td>
<td>0.35 eV – 35 eV</td>
</tr>
<tr>
<td>Simpson et al. (1974)</td>
<td>Transmission</td>
<td>0.5 eV – 1 keV</td>
</tr>
<tr>
<td>Berreth et al. (1970)</td>
<td>Transmission</td>
<td>0.008 eV – 25.6 eV</td>
</tr>
<tr>
<td>Cote et al. (1959)</td>
<td>Transmission</td>
<td>0.0014 eV – 15.44 eV</td>
</tr>
<tr>
<td>Weston et al. (1985)</td>
<td>Capture</td>
<td>258 eV – 92.1 keV</td>
</tr>
<tr>
<td>Wisshak et al. (1983)</td>
<td>Capture</td>
<td>5 keV – 250 keV</td>
</tr>
<tr>
<td>Mendoza et al. (2013)</td>
<td>Capture</td>
<td>0.7 eV – 2.5 keV</td>
</tr>
<tr>
<td>Jandel et al. (2009)</td>
<td>Capture</td>
<td>8 eV – 250 keV</td>
</tr>
<tr>
<td>Hori et al. (2009)</td>
<td>Capture</td>
<td>0.01 eV – 400 eV</td>
</tr>
<tr>
<td>Kimura et al. (2012)</td>
<td>Capture</td>
<td>RP of the 1.35 eV resonance</td>
</tr>
<tr>
<td>Alekseev et al. (2011)</td>
<td>Fission</td>
<td>RP below 17 eV</td>
</tr>
</tbody>
</table>

In order to reproduce the Ohta et al. resonance integral value given in Table 11, the latest releases of ENDF/B and JENDL (ENDF/B-VII.1 and JENDL-4.0) have increased the $\Gamma_\gamma$ and $\Gamma_n$ of the strongest $^{243}\text{Am}$ resonance at 1.35 eV, which is the main contribution to the mentioned integral (70-80%). Such a modification poses some problems, because the values of $\Gamma_n$ provided by Bellanova et al. Cote et al. and Alekseev et al. are even lower than the ones provided by Simpson et al. and Berreth et al. The significant increase of $\Gamma_\gamma$ leads to a cross-section, which is not compatible with the Simpson et al. data. On the other hand, the $\Gamma_\gamma$ and $\Gamma_n$ values provided by Kimura et al. are between the new and the old evaluated ones, but they were obtained from the analysis of the $^{243}\text{Am}$ impurities in a $^{244,246}\text{Cm}$ capture measurement, thus requiring the use of an evaluated $^{243}\text{Am}$ cross-section.

Concerning the rest of the resolved resonance region, the capture cross-section measured by Mendoza et al. at n_TOF suggests that the evaluations available in 2013 are underestimating the $^{243}\text{Am}$ capture cross-section up to 12% (ENDF/B libraries) or 25% (Maslov evaluation, rest of the libraries) as it is shown in Figure 8. In addition, the experimental data by Mendoza et al. allow extending the resolved resonance region up to 400 eV.
The unresolved resonance region ranges from 250 eV up to 40-42 keV in the present evaluations. They rely only on the differential data sets of Weston et al. hereafter referred as Weston I and Weston II, and the data by Wisshak et al. (see Table 12 and references therein for details). The Wisshak et al. data set is around 10% below the Weston et al. data in the neutron energy range where they overlap. Up to 5 keV, only the two datasets provided by Weston et al. are available, which differ below 2 keV. As can be observed in Figure 9, the present evaluations are closer to the Weston II data set. However, the Mendoza et al. data (n_TOF) are consistent with the Weston I data set. This suggests that the present evaluations underestimate the cross-section in this energy region.

In addition, none of the libraries but ENDF/B-VII.1 and JEFF-3.0 reproduces the PROFIL-1 integral experiment performed at the PHENIX fast reactor [140-142]. The results of PROFIL-1 indicate that the evaluated capture cross-sections are too low in the fast region. Indeed, the ENDF/B-VII.1 library [142] is compatible with PROFIL-1 because it has been adjusted ad hoc, by increasing it around 15% above 10-20 keV. A consequence of this adjustment is the loss of compatibility with the experimental differential data.
An alternative mechanism has been proposed in [136] for obtaining a capture cross-section, which is compatible with the differential data sets and which reproduces the PROFIL-1 experiment.

As a summary, it can be concluded that the $^{245}$Am($n,\gamma$) cross-section presents its largest uncertainties at thermal energies, at the strongest resonance energy of 1.35 eV and in the fast range for reactor applications. The Mendoza et al. experimental results suggest some changes in the present evaluated libraries in the resolved energy region and in the range between 400 eV and 2.5 keV. The experimental results of Jandel et al. and Hori et al. could further reduce some of the mentioned uncertainties.

$^{237}$Np($n,\gamma$)

The thermal capture cross-section of $^{237}$Np is important for nuclear applications and several recent measurements have been performed. While some thermal data were obtained from pile oscillation measurements [143] or estimated from transmission [144] or capture time-of-flight measurements [145] [146], many of the measurements are based on the activation technique. By irradiating $^{237}$Np in a known neutron flux, the quantity of produced $^{238}$Np is determined by either $\alpha$- or $\gamma$-spectroscopy using the decay chains of $^{235}$Np and $^{238}$Np [147–154]:

$$^{237}\text{Np} \left( 2.1 \times 10^6 \text{ y} \right) \rightarrow ^{233}\text{Pa} \left( 27 \text{ d} \right) \rightarrow ^{233}\text{U} \left( 1.6 \times 10^5 \text{ y} \right)$$

$$^{238}\text{Np} \left( 2.1 \text{ d} \right) \rightarrow ^{238}\text{Pu} \left( 87.7 \text{ y} \right) \rightarrow ^{234}\text{U} \left( 2.5 \times 10^5 \text{ y} \right)$$

With gamma spectroscopy, the 312 keV gamma-line of $^{233}$U* in the natural decay chain of $^{237}$Np can be compared to the 984 keV line from the decay of $^{238}$Np formed after neutron capture. With alpha spectroscopy, the alpha particles of $^{237}$Np are compared to the alphas of $^{234}$U present in the decay chain of $^{238}$Np which are well separated from those of $^{237}$Np.

The measurements based on $\gamma$-spectroscopy depend largely on the accuracy of the gamma-ray intensities used. A new measurement made by Harada et al. [152] determined also a correction of the values for the $\gamma$-intensity based on $\alpha$ spectroscopy allowing a correction for the previous $\gamma$-ray based measurements in [149 –151]. Letourneau et al. [153] reported later a $\gamma$-ray based measurement with a different correction for the $\gamma$-intensities, which would alter the previous measurements. Finally, Genreith et al. [154] reported a recent measurement on a cold neutron guide in absence of epithermal neutrons, and quoting two values, one based on neutron-induced activation $\gamma$-rays and the other on prompt neutron capture $\gamma$-rays relative to $^{197}$Au($n,\gamma$). Again, the available data on the gamma-ray intensities and the corrections for gamma-ray absorption determine the quality of the final results.

Not only the $\gamma$-ray intensities, but also the details of the neutron flux used for irradiation are important to determine the cross-section at 2 200 m/s. The first resonance of $^{237}$Np at 0.5 eV makes it difficult to shield or correct for epithermal neutrons.

Nevertheless, the overall agreement of the different datasets, which are shown as originally reported in Figure 10, is satisfactory given the uncertainties. Analysis techniques after activation, different from $\alpha$- or $\gamma$-spectroscopy, for example based on mass spectroscopy, could possibly help reduce the uncertainties and improve the thermal neutron cross-section.
Figure 10: Summary of $^{237}$Np($n,\gamma$) thermal cross-section measurements

For resolved resonances, several measurements were used to extract resonance parameters. The resonance parameters were first extracted for the first three resonances from the transmission measurement by Smith et al. [144]. Paya [155] measured fission and transmission. Capture, transmission and elastic scattering were measured by Mewissen [156]. The capture yield was measured by Weston and Todd [145] and transmission and fission were measured by Auchampaugh et al. [157]. More recently, transmission was measured by Gressier [158] at GELINA. From KURRI, three capture measurements are reported. Kobayashi [159], using both a TOF experiment and a lead slowing-down spectrometer, Shcherbakov [160] measuring capture with a total energy BGO detector, and Mizumoto [161] using a $4\pi$ germanium array. Esch [146] reported a capture measurement from DANCE. Finally, Guerrero et al. [162] measured capture at n_TOF. Since cross-sections in the resolved energy region are represented by resonance parameters, the most appropriate way to evaluate these datasets into a consistent set of resonance parameters would be by a simultaneous R-matrix analysis, including realistic resolution functions, Doppler broadening models, and other experimental considerations, and an average analysis for the unresolved energy range above about 500 eV.

At higher energies, a list of activation measurements in various fast neutron fluxes (reactors, spallation sources, Van de Graaff accelerators and even an underground explosion) [163–173] contribute to our knowledge of the capture cross-section measurement of $^{237}$Np and can validate cross-section model calculations for an evaluation.

3.2 Current status of the cross-section for the $^{10}$B($n,\alpha$) reaction

The $^{10}$B($n,\alpha$) reaction is listed in the SG26 report [1] for reasons of neutron removal for the different reactor types up to energies of 1.3 MeV. The $^{10}$B($n,\alpha$) reaction has long been given particular attention because of its important role as a neutron cross-section standard [174].

The reaction $^{10}$B($n,\alpha$) stands for the sum of the two standard reactions $^{10}$B($n,\alpha_0$)$^7$Li (Q-value 2.792 MeV) and $^{10}$B($n,\alpha_1$)$^7$Li (Q-value of 2.314 MeV). The evaluated cross-sections [175] at 25.3 meV are 3839.4 b for $^{10}$B($n,\alpha$) with partial cross-sections of 241.3 b and 3598.2 b for $^{10}$B($n,\alpha_0$) and $^{10}$B($n,\alpha_1$), respectively. The branching ratio $^{10}$B($n,\alpha_0$/$^{10}$B($n,\alpha_1$) is about 0.06 at thermal energies.
Different detection techniques have been employed to determine the two reactions or their branching ratio. Alpha and $^7\text{Li}$ particle spectroscopy can be used for both reactions, while the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction is well suited for gamma spectroscopy techniques because of the emission of the 478 keV gamma ray. At thermal energies, $^{10}\text{B}(n,\alpha)$ can be deduced from the total cross-section data corrected by scattering.

Since the $^{10}\text{B}(n,\alpha)$ reaction is in the resolved energy region up to several MeV, it can be described by the R-matrix formalism. Therefore, several other reactions provide crucial information, allowing a significant constrain on the evaluation of the $^{10}\text{B}(n,\alpha)$ cross-section and on the reduction of uncertainties. These reactions concern all reactions leading to the compound system $^{11}\text{B}$, not only neutron-induced reactions such as $^{10}\text{B}$ total, differential, integrated and polarised scattering $^{10}\text{B}(n,n)$ and $^{10}\text{B}(n,\alpha)$ reactions, but also inverse reactions like $^7\text{Li}(n,\alpha)$ [176–179]. The angular distribution of the $^{10}\text{B}(n,\alpha)$ and $^{10}\text{B}(n,\gamma)$ reactions are isotropic at low energy but become anisotropic above about 100 keV. High-quality angular distribution data like the recent measurements by Hambsch and Ruskov [180] are a crucial part of the response of detector systems for $^{10}\text{B}(n,\alpha)$ measurements. In addition, they are important to constrain a global combined R-matrix fit.

The recent standard evaluation [175] extends the $^{10}\text{B}(n,\alpha)$, as well as its components $^{10}\text{B}(n,\alpha)_0$ and $^{10}\text{B}(n,\alpha)_1$ as a standard in the energy range between 0.0253 eV and 1 MeV. It also contains the list of datasets for $^{10}\text{B}$ from the so-called GMA database, which are used for the evaluation.

For the $^{10}\text{B}(n,\alpha)$ reaction, only a few cross-section measurements are reported at thermal energies. Additional measurements concern natural boron consisting of the two isotopes $^{10}\text{B}$ (19.8%) and $^{11}\text{B}$ (80.2%).

The total cross-section at thermal energy consists nearly entirely of the $^{10}\text{B}(n,\alpha)$ reaction, but a small fraction of about 0.07% is due to other reactions, such as $^{10}\text{B}(n,\gamma)$ [181] [182], $^{10}\text{B}(n,p)^{11}\text{B}$ [183], and the tritium producing reactions $^{10}\text{B}(n,T)^{8}\text{Be}$ and $^{10}\text{B}(n,\alpha+T)\alpha$. The coherent scattering amplitude and cross-section have been reported as well [184] [185].

The $^{10}\text{B}(n,\alpha)$ thermal cross-section has been measured as the absorption cross-section by means of a transmission experiment [186–188] or sometimes in pulsed experiments with fast neutrons by measuring the decrease of the neutron flux in water, with and without $^{10}\text{B}$ [189] [190]. The total cross-section measurement of Diment et al. [191] could be corrected for the scattering contribution and gives an additional $(n,\alpha)$ thermal data point. These datasets form the only thermal measurements of $^{10}\text{B}(n,\alpha)$.

More datasets are available for natural boron. Since $^{11}\text{B}$ has a total cross section of only about 5 b at thermal energies, measurements on natural boron are important sources of information for the thermal cross-section of $^{10}\text{B}$. Most of these measurements use the same techniques as for $^{10}\text{B}$. There are three ratio measurements of natural boron absorption relative to hydrogen capture [192–194] and one ratio measurement relative to $^{55}\text{Mn}$ [195]. Figure 11 shows the values and the standard deviations reported for $^{nat}\text{B}(n,\alpha)$ and $^{10}\text{B}(n,\alpha)$ in chronological order.
At higher energies, above thermal energies, several total cross-section measurements exist. At thermal and epithermal energies the \((n,\alpha)\) cross-section is usually obtained by subtracting the scattering cross section. Although below 1 MeV no other reaction channels than \((n,\alpha)\) and elastic scattering contribute significantly to the total cross-section, the ratio of the \((n,\alpha)\) to elastic cross-section decreases rapidly from roughly 1500 at 25 meV to 1 at 50 keV. Scattering measurements from Lane et al. [196] and earlier work, together with total cross-section measurements as in [197] and earlier work, or data from the inverse reaction \(^7\text{Li}(\alpha,n)\) [176-179] are therefore all necessary ingredients for a complete analysis of the \(^{10}\text{B}(n,\alpha)\) cross-section in a consistent R-matrix analysis.

Outside the \(^{10}\text{B}(n,\alpha)\) standard region above 1 MeV, a rather limited number of datasets concern direct \((n,\alpha)\) measurements. Bichsel and Bonner [198] reported measurements with a \(^{10}\text{BF}_3\) counter. Friesenhahn et al. [199] [200] measured also with a \(^{10}\text{BF}_3\) detector. Zhang et al. [201] used a gridded ionisation chamber. Also Giorginis and Khryachkov [202] used a gridded ionisation chamber and used a data acquisition based on waveform digitisers, allowing an improvement of the event selection. Finally, Zhang et al. [203] reported a measurement using a gridded ionisation chamber. These few datasets are still largely inconsistent with each other. This is reflected in the considerable differences
between the ENDF/B-VII.1 (and JEFF-3.1.2) and JENDL-4.0 evaluations in the MeV region, in
counter to the excellent agreement seen up to about 100 keV, as shown in Figure 12
together with experimental data from EXFOR.

Figure 12: The different experimental datasets for $^{10}$B(n,$\alpha$) cross-section together with the
evaluated cross-section from both ENDF/B-VII.1 and JENDL-4.0

The analysis of a new measurement performed at JRC-IRMM by Bevilacqua et al. is
ongoing with preliminary results presented at the ND2013 conference [205]. Also for this
experiment, a gridded ionisation chamber has been used with $^{235}$U(n,f) as reference.

The recent evaluations for $^{10}$B have full covariance information due to modern
evaluation techniques. For ENDF/B-VII.1, the evaluated uncertainties are very low as a
result of the combined analysis where only reported uncertainties were used and vary
from 0.1% at thermal to about 1.5% in the MeV region, as shown in Figure 13. This library
also mentions extended uncertainties, which are much higher, and which would make
the evaluation consistent with the considered datasets. For JENDL-4.0, the evaluation
contains higher and probably more realistic uncertainties, also shown in Figure 13.
Nevertheless, future high-quality measurements above 100 keV, using sophisticated
signal analysis techniques like in [202] could help pin down inconsistencies in existing
datasets and lead to realistic but improved uncertainties in a future evaluation.

Figure 13: The relative standard deviation on the $^{10}$B(n,$\alpha$) cross-section from ENDF/B-VII.1 and
JENDL-4.0 (data plotted with JANIS [204]).
3.3 Current status on inelastic cross-section measurements

3.3.1 On-going work

To the best of our knowledge on-going work for inelastic scattering on major actinides is limited to the collaboration led by the Strasbourg group [206-208]. In this work, $^{235}$U, $^{238}$U and $^{232}$Th have been measured using the (n,n'γ)-technique at the IRMM GELINA facility and there are firm plans for measuring $^{233}$U with the same setup. The results for $^{235}$U have been published [209] and the data analysis for the other isotopes is in progress. The gamma-ray production cross-sections should be compared to earlier work, notably from the Los Alamos group using the GEANIE array (e.g. [210] [211]), the TUNL group [212] and the Lowell group [213]. The main purpose of these cross-sections is to serve as an accurate benchmark for model calculations. In most cases, for actinides, missing transitions (due to overlap with fission products or natural activity gamma-rays) and large conversion coefficients for low-energy transitions preclude complete results beyond a given excitation energy. However, in the favourable case of $^{232}$Th, the data are complete up to 774 keV excitation energy. Nevertheless, the accuracy for the important 49 keV transition is compromised by the above mentioned effects.

3.3.2 Recent and earlier work

1990-present: Recent $^{238}$U(n,n'γ) data were published by the TUNL group [212]. It is not clear if they have a continuing experimental programme. A collaboration between LANL and LLNL carried out measurements for major actinides ($^{235,238}$U) with the (n,n'γ)-technique using the GEANIE array at the LANSCE neutron time-of-flight facility between 1995 and 2005 (see [210] [211] and references therein). From 1990 to 1999 Subgroup 4 [214] of the NEA WPEC stimulated a considerable experimental effort regarding inelastic scattering on $^{238}$U (Baba et al. at Tohoku University [215] [216] and references therein, Kornilov et al. at Obninsk [217], Moxon et al. at IRMM [218] and Plompen et al. also at IRMM [219]). All works for this subgroup focused on the time-of-flight technique ((n,n')-technique) and the detection of the emitted neutrons at one or several well defined angles. The work of Moxon et al. was special in this respect since it involved a filtered beam at the GELINA time-of-flight facility. The other experiments were carried out using quasi mono-energetic neutrons from the $^7$Li(p,n), $^3$H(p,n) or $^2$H(d,n) reactions. In parallel, measurements of cross-sections were made for $^{233}$Th by Miura et al. [216] [220], Smith and Chiba at ANL [221], for $^{238}$U by Kornilov and Kagalenko [217], and for $^{239}$Pu by Staples et al. [222] and by Yue et al. [223] at the University of Lowell. At the University of Kentucky, the spin and parity assignment of the 1414 keV excited state was studied via angular distributions of the emitted gamma-rays produced by inelastic neutron scattering and some un-normalised γ-ray intensities were reported in [224].

1950-1990: In the period prior to 1990 the largest number of measurements for inelastic scattering on the major actinides concern (n,n') experiments at pulsed quasi mono-energetic neutron sources. The better-known laboratories involved were Bruyères-le-Châtel [225], Argonne National Laboratory [226], the University of Lowell [227], IPPE Obninsk [228], Los Alamos National Laboratory [229], Aldermaston [230], Oak Ridge National Laboratory [231] and the Institute of Atomic Energy in Beijing [232]. Gamma-ray detection or (n,n'γ) experiments were carried out at both pulsed quasi mono-energetic (Dave et al. [233] at Lowell) and a pulsed white neutron source (Olsen et al. at Oak Ridge National Laboratory [234], Voss et al. [235] at the Forschungszentrum für Kernphysik, Karlsruhe). Filtered beam experiments were carried out using reactor beams (L.L. Litvinskiy et al. Kiev [236]) and pulsed accelerator sources (R.R. Winters et al. ORNL [237]). The sphere transmission technique, promoted by Bethe and Beyster [238]

1 Only one reference is given for each of these laboratories. Please check references therein and other sources (EXFOR) for additional works.
was applied up to 1963 (Allen et al. LANL [239], M.H. McTaggart and H. Goodfellow, AWE [240]). It provides a measurement of the non-elastic cross section with reasonably good accuracy. Part of the reason for its limited use probably lies in the mass of the sphere that is required.

### 3.3.3 Status and outlook

It has been shown in several places that the scatter among the results from the different experiments is relatively large. In part, this may reflect progress in measurement technique, such as the optimisation of sample geometry, shielding, detector efficiency calibration, method of normalisation, and handling of background. In part, this reflects the fact that the results shown most often are derived data. In particular, level cross-sections and total inelastic cross sections derived from neutron time-of-flight spectra obtained at a number of angles, require important corrections to go from the measurement results to the reported data. For instance, the fission spectrum must be subtracted and elastic scattering must be separated from inelastic scattering in order to report elastic scattering and inelastic scattering differential cross-sections. Furthermore, one has to integrate over angle and obtain the yield of neutrons at low-emission energy by an extrapolation to account for the detection threshold. Most measurements do not resolve the states of the ground state rotational band from the elastic scattering peak and therefore report differential scattering data for “pseudo-elastic” scattering implying a sum over the contributions from the 0+ - 6 (8+)

It is assumed that the double differential data are nevertheless very valuable and may provide more accurate information than the derived experimental data uncertainties suggest. If used properly it may be that comparison between theory and experiment leads to valuable improvements for modelling of inelastic scattering. It is therefore worthwhile to review the available data in order to check whether the original double differential data are still accessible, what is really reported in the paper, and how the derived data depend on assumptions about the fission spectrum and elastic scattering. Such a review should be aware of the fact that it is difficult to get a comprehensive overview of all pertinent work using the available EXFOR browsers from IAEA/NNDC or NEA. To get a complete overview it is recommended to look at several recent publications (e.g. P. Young’s report on actinide evaluations for ENDF/B-VII in Nuclear Data Sheets, the JENDL-4 evaluation report, etc.).

Finally, it is crucial to stimulate experimentalists to measure this important cross-section on major actinides in the interest of nuclear applications. The possible use of new techniques and new facilities should be promoted to allow improvements over past results.

### 3.4 Experiments on prompt fission neutrons in inverse kinematics

Important progress in fission experiments has been achieved by studying the fission of high energetic projectiles induced by interactions with target nuclei [241-243]. This “inverted” kinematical condition allows studying fission of short-lived nuclei that cannot be provided as target material. Fragmentation of relativistic 238U gives access to a large number of projectile fragments with mass number A ≤ 238 and Z ≤ 92 [241]. Isotopes of Z=93 and isotones with N=147, but always A ≤ 238, are accessible by charge-changing reactions [244]. Transfer reactions of 238U with an energy slightly above the Coulomb barrier produce nuclei in the vicinity of 238U, which can also be heavier than A=238 [242].

Comparable kinematical conditions may be offered in the future at CERN ISOLDE with post-accelerated radioactive beams produced by spallation of 238U with a diversity of fissioning systems similar to projectile fragmentation [245]. Moreover, in inverse kinematics the possibilities for identifying the fission products are considerably improved.
by their appreciably higher energies if compared to traditional experiments in normal kinematics. The atomic numbers of all fission fragments can unambiguously be determined in an ionisation chamber [246], and their masses can be deduced from the deflection in a dipole magnet [247] [248] and a time-of-flight measurement. These new approaches also offer new possibilities for experimental studies on prompt-neutron emission in fission.

3.4.1 Direct detection of prompt neutrons

Neutron detectors with kinematical information have been developed, specifically suited for GANIL energies [249] and for SIS energies [250]. The characteristics of scintillation neutron detectors at energies up to 12 MeV have been studied in [251]. Others provide the neutron multiplicity [252] [253]. The detection of neutrons emitted from fissioning systems at GANIL energies is not much different from experiments in normal kinematics. A major problem, however, results from the large amount of matter provided by the magnetic spectrometer that is needed for identification of the fission products, which leads to substantial neutron absorption and scattering. Thus, direct kinematical neutron detection is not a promising option for this kind of experiments. The neutron detector LAND of GSI has been used specifically in fission experiments (e.g. [254] [255]) with a relativistic $^{238}\text{U}$ beam. It has an active area of 2 m times 2 m, a high detection probability above 90%, a position resolution of 7 cm, and a time resolution of 460 ps [256]. The ALADIN dipole allows free passage of the fission neutrons to the LAND detector that is mounted about 10 m behind the target, because the neutrons are emitted in a narrow cone in forward direction. The conditions will be similar in the future R3B set-up [257], where a new dipole magnet (GLAD) and a new neutron detector will be available. Thus, the complete identification of both fission products in atomic number $Z$ and mass number $A$ can be achieved together with the detection of all neutrons emitted from the fissioning system and from the fission fragments. The kinematical properties can be determined with an uncertainty given by the resolutions in time and position of the LAND detector. However, high multiplicities lead to an overlap of the signals of the individual neutrons.

3.4.2 Indirect determination of mean neutron multiplicities

The mean multiplicity of prompt fission neutrons as a function of fragment mass can also indirectly be determined by measuring the masses of the fission fragments before and after neutron emission, see e.g. [258]. The traditional method consists of a double-energy, double-time-of-flight measurement. In inverse kinematics, the post-neutron mass of one or both fission products can unambiguously be determined with the magnetic spectrometer, e.g. VAMOS or ALADIN, respectively GLAD. The pre-neutron mass can be obtained from a measurement of the emission angles of the fission products. The ratio of the momentum components transversal to the beam direction is proportional to the inverse of the mass ratio. This defines the masses of the fission fragments if the mass of the fissioning system is known. In transfer reactions, the compound nucleus is known from a measurement of the light transfer product. Multichance fission, i.e. neutron emission prior to fission, complicates the situation. Application to electromagnetic-induced fission in a heavy target material at relativistic energies suffers from a background of nuclear-induced events that can only partly be suppressed on an even-by-event basis by the condition that all protons of the fissioning system should be present in the fission fragments [241]. This problem will be solved in fission induced by tagged photons in a heavy-ion - electron collider ring [259]. If isotropic neutron emission in the respective fragment frame is assumed, it does not change the emission angle on the average. The resolution is determined by the precision with which the fragment emission angles are determined, by the recoil of the emitted neutrons and by the angular straggling of the projectile in the first section of the target and the fission fragments in the remaining target section. An additional uncertainty originates from the
emittance of the primary beam or, if the individual projectiles are tracked, from the resolution of the dedicated position detectors.

The calibration of the pre-neutron-mass-ratio determination is based on a direct geometrical measurement. The resolution obtained for a single event is not crucial: It is the uncertainty of the mean value of the pre-neutron fragment mass for a given post-neutron fission-product mass that determines the uncertainty of the mean prompt-neutron multiplicity.

3.4.3 Outlook

Experiments in inverse kinematics considerably extend the diversity of fissioning systems for experimental studies. Measurements on prompt neutrons are feasible. Direct detection of all fission neutrons is possible with relativistic beams, since the large dipole magnet (e.g. ALADIN or GLAD) provides free passage for the neutrons. Indirect determination of mean prompt-neutron multiplicities as a function of the fission-product mass is possible when the mass of the fissioning nucleus is known. This information is available in electromagnetic-induced fission in the case of first-chance fission, and it is accessible in transfer-induced fission.
4. Recommendation of collaborative path forward to meet the needs

To obtain accurate and reliable nuclear data satisfying the target accuracy, measurements with the precision equivalent to the target accuracy are necessary together with appropriate descriptions of data reduction and uncertainty evaluation in a research report. Furthermore, to verify the nuclear data and their uncertainties, it is not enough to repeat simply the same kind of measurements, but it is recommended to proceed to independent measurements, for example, using different measurement principles, facilities, detectors, samples, experimenters, which may affect the experimental values. Especially, combined use of differential and integral measurements is recommended to achieve the target accuracy. These efforts are indispensable to eliminate unrecognised systematic effects as much as possible.

The difficulty of appropriate descriptions of data reduction and uncertainty evaluation for each experiment is sometimes caused by the limitation in the number of pages on research reports, in many cases, Journal articles. Even in the case that there is a technical report that describes the details of the detectors and/or the facilities, it does not include all the procedures for data reductions and uncertainty evaluations in detail. These procedures depend on a laboratory or even an experimenter. This is one of the negative factors that make the evaluations difficult. Comprehensive and unified guides should be prepared for each experimental method in order to describe the details of the data reduction and uncertainty evaluation. See [260] [23].

As discussed in Section 3, the difficulties of measurements and the origins of unrecognised systematic effects are different for each reaction and nucleus. However, the collective knowledge of known systematic effects can be used for the case of other nuclear data. Comprehensive knowledge of these systematic effects for each measurement type (capture, fission, inelastic, and ν) will be a valuable base for both experimenters and evaluators, and should be summarised as a series of research reports. The recent review paper on capture [23] is a good example.

To guarantee independence of experiments, international collaborations are indispensable, making sure not to be disturbed by the “Phase locking effect” pointed out by Lawrence; the phrase means that once one’s measurement agreed with a celebrated expert’s measurement, one stops doubting one’s measurement and misses to find unrecognised systematic effect. In order to demonstrate the effectiveness of such collaborations, an appropriate framework should be established, where serious scientific discussions are possible.

Nuclear data together with their uncertainties should be revised periodically by an appropriate international committee [261], such as the Committee on Data for Science and Technology (CODATA) [262] in the field of fundamental physical constants.
5. Summary

To meet the requirement of accurate nuclear data for developing advanced nuclear systems, pertinent efforts in the fields of experiments and evaluations are still required and indispensable. As described in Section 3, there are striking technical advancements in nuclear data measurement methods. For example, high-intensity-pulsed neutrons generated by spallation reaction at CERN in Europe, LANSCE in USA, and J-PARC in Japan are available to obtain high-precision neutron TOF data. Finer corrections of traditional techniques are also possible by using recent Monte Carlo simulation techniques or refining existing data reduction codes such as REFIT, SAMMY etc. New concept of detectors and innovative methods using inverse reactions have also been developed and applied for nuclear data measurements. By using these state-of-art techniques, further improvements of nuclear data accuracy are expected.

It is understood that the experimental result is the best estimate of the value of the measurement, and that all components of uncertainty contribute to the total uncertainty. However, some systematic effects are sometimes unrecognised and not discussed in published papers. It should be noted that only the known systematic effects are corrected and took into account in the total uncertainty. Some recommendations of collaborative path forward to meet the needs were summarised in Section 4.

In order to obtain accurate nuclear data, it is important to measure nuclear data precisely and identify the unrecognised systematic effects as much as possible. Double-check experiments are indispensable to verify the results. International collaborations are effective in guaranteeing the independence of experiments. In order to demonstrate the effectiveness of such collaboration, an appropriate framework should be established, where serious and detailed scientific discussions are possible.
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Appendix A: Graphical presentation of EXFOR data

A selection of reactions were shown in the main part of the document, others are presented in this appendix.

The experimental nuclear reaction data library EXFOR [1], actively kept up to date by the International Network of Nuclear Reaction Data Centres (NRDC) [2], is the reference tool to access experimental nuclear data. In order to easily assess the available experimental data on a particular reaction, a summary of the available data sets for the reaction is indispensable.

We have tempted to produce a concise graphical presentation of each of the reactions relevant for this subgroup. In earlier versions the used entries were based on the C4 summary table of the EXFOR library, which is regularly updated and made available [3]. To have more flexibility we then used directly the data from the mySQL database, also available from the IAEA [4]. This database contains many of the records of the initial EXFOR file. The information of this database was processed to extract for each reaction a number of relevant items to construct the summary plots in an automated way.

We used the programming language Perl [5] to construct the SQL queries, retrieve the results from the database, uncompress the data records on the fly where necessary, and process them to generate the plots using the scriptable plotting package Gnuplot [6].

A match is determined by an EXFOR-formatted reaction string, composed of the entrance channel, for example ‘5-B-10(n,’, the exit channel, for example ‘g), ‘tot)’ or lumping several exit channels together written as for example as ‘a|abs)’. Also part of the match is the measurement quantity, EXFOR-formatted, or a list of quantities separated by a ‘|’, like ‘sig)’ or ‘sig|wid|zi)’. Each matching entry is then displayed as a bar on the energy region it covers. The energy limits are slightly enlarged in order to obtain a visible representation of entries at a single energy, like thermal values. The entries are sorted by year to show a chronological history of experiments. The number of entries per plot is limited and the total number of entries is spread over several plots if needed. No additional selection is performed on the entries. Next to the bar the following basic information is displayed:

- Year;
- Author information;
- EXFOR accession number;
- Number of data points $n$;
- Part of the reaction string $r$;
- Average uncertainty $\frac{\Delta \bar{X}}{\bar{X}} \times 100\%$, also represented by the colour of the bar.
Table A.1: The colour scale used for the total average uncertainty for the data sets

<table>
<thead>
<tr>
<th>Colour</th>
<th>Uncertainty range (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Black</td>
<td>unavailable</td>
</tr>
<tr>
<td>Blue</td>
<td>0.0 &lt; ( \sigma_x ) &lt; 2.0</td>
</tr>
<tr>
<td>Green</td>
<td>2.0 &lt; ( \sigma_x ) &lt; 5.0</td>
</tr>
<tr>
<td>Yellow</td>
<td>5.0 &lt; ( \sigma_x ) &lt; 10.0</td>
</tr>
<tr>
<td>Orange</td>
<td>10.0 &lt; ( \sigma_x ) &lt; 20.0</td>
</tr>
<tr>
<td>Red</td>
<td>( \sigma_x ) &gt; 20.0</td>
</tr>
</tbody>
</table>

The colour of the bar represents the reported average uncertainty on a colour scale, as shown in Table 1. The uncertainty value is also reported in the figures. When no uncertainty information is available, the bar colour is set to black. For single data point the uncertainty is straightforward. For a dataset consisting of many points \( x_i \) it is less evident to summarize the uncertainty information in a single number. Also due to the rather flexible way of storing data columns in EXFOR, it is not always trivial to retrieve the columns corresponding to data and uncertainties in an automated way. Also the interpretation of the quoted uncertainties (for example only due to counting statistics) may depend on each data set. Despite these considerations, the column with data points \( x_i \) and the column with uncertainties \( \Delta x_i \) is searched for and extracted, and then used to calculate an average uncertainty \( \sigma_x \) defined as:

\[
\sigma_x = \frac{\sum_i |\Delta x_i|}{\sum |x_i|}.
\]

The results are given in the figures on the following pages for each of the nuclei considered. The reactions and nuclei are extracted from the NEA/WPEC-26 subgroup report [7], where nuclear data needs are identified for a number of isotopes, reactions and energy ranges relevant for 8 different reactor systems. These systems are taken from the Generation-IV list [8] and consist of a very high-temperature reactor and several types of fast reactors, all based on the uranium cycle. The inventory of existing experimental data in EXFOR has been summarized for the isotopes and reactions taken from the SG26 report, and the energies ranges have been extended to cover all available data sets in order to have a complete view of the existing data. Three items (O, C, Pb) were in some cases originally listed as elements. In these cases we included in the EXFOR overview the search results for data on the element (indicated by a 0, \(^{12}\)C, \(^{16}\)O and \(^{206,207,208}\)Pb) as well as on the most abundant natural isotopes (\(^{12}\)C, \(^{16}\)O and \(^{206,207,208}\)Pb). Also the reaction listed originally at one point as “scattering” has been included in both “elastic” and “inelastic” scattering. Table 2 gives an overview of the included isotopes and reactions.

The resulting match of subentries in the plots can be of help in getting an overview of datasets for a particular reaction. Nevertheless, the plots still need careful examination in order to select the relevant datasets.
Table A.2: Overview of the isotopes and reactions as summarised from the SG26 report

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<th>v</th>
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References


Figure A.1: Summary of the 96 EXFOR entries for the nuclide $^{241}$Am, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 3
Figure A.2: Summary of the 96 EXFOR entries for the nuclide $^{241}\text{Am}$, and reactions $(n,g)$, $(n,\text{abs})$, for entries matching the quantities sig, wid, ri, par; part 2 of 3

Am-241 $(n,g|\text{abs})$ sig|wid|ri|par

Figure A.3: Summary of the 96 EXFOR entries for the nuclide $^{241}\text{Am}$, and reactions $(n,g)$, $(n,\text{abs})$, for entries matching the quantities sig, wid, ri, par; part 3 of 3

Am-241 $(n,g|\text{abs})$ sig|wid|ri|par
Figure A.4: Summary of the 77 EXFOR entries for the nuclide $^{243}$Am, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Am-243 (n,g|abs) sig|wid|ri|par

Figure A.5: Summary of the 77 EXFOR entries for the nuclide $^{243}$Am, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Am-243 (n,g|abs) sig|wid|ri|par
Figure A.6: Summary of the 22 EXFOR entries for the nuclide $^{244}\text{Cm}$, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Cm-244 (n,g|abs) sig|wid|ri|par

Figure A.7: Summary of the 60 EXFOR entries for the nuclide $^{56}\text{Fe}$, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Fe-56 (n,g|abs) sig|wid|ri|par
Figure A.8: Summary of the 60 EXFOR entries for the nuclide $^{56}$Fe, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Fe-56 (n,g|abs) sig|wid|ri|par

Figure A.9: Summary of the 40 EXFOR entries for the nuclide $^{58}$Ni, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Ni-58 (n,g|abs) sig|wid|ri|par
Figure A.10: Summary of the 87 EXFOR entries for the nuclide $^{237}$Np, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 3

Figure A.11: Summary of the 87 EXFOR entries for the nuclide $^{237}$Np, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 3
Figure A.12: Summary of the 87 EXFOR entries for the nuclide $^{237}$Np, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 3 of 3

Figure A.13: Summary of the 22 EXFOR entries for the nuclide $^6$Pb, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1
Figure A.14: Summary of the 34 EXFOR entries for the nuclide $^{206}$Pb, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pb-206 (n,g|abs) sig|wid|ri|par

Figure A.15: Summary of the 27 EXFOR entries for the nuclide $^{207}$Pb, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pb-207 (n,g|abs) sig|wid|ri|par
Figure A.16: Summary of the 42 EXFOR entries for the nuclide $^{208}$Pb, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Pb-208 (n,g|abs) sig|wid|ri|par

Figure A.17: Summary of the 42 EXFOR entries for the nuclide $^{208}$Pb, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Pb-208 (n,g|abs) sig|wid|ri|par
Figure A.18: Summary of the 19 EXFOR entries for the nuclide $^{238}\text{Pu}$, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pu-238 (n,g|abs) sig|wid|ri|par

Figure A.19: Summary of the 61 EXFOR entries for the nuclide $^{239}\text{Pu}$, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Pu-239 (n,g|abs) sig|wid|ri|par
Figure A.20: Summary of the 61 EXFOR entries for the nuclide $^{239}\text{Pu}$, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Pu-239 (n,g|abs) sig|wid|ri|par

![Graph Image]

Figure A.21: Summary of the 52 EXFOR entries for the nuclide $^{240}\text{Pu}$, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Pu-240 (n,g|abs) sig|wid|ri|par

![Graph Image]
Figure A.22: Summary of the 52 EXFOR entries for the nuclide $^{240}$Pu, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Pu-240 (n,g|abs) sig|wid|ri|par

Figure A.23: Summary of the 28 EXFOR entries for the nuclide $^{241}$Pu, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pu-241 (n,g|abs) sig|wid|ri|par
Figure A.24: Summary of the 38 EXFOR entries for the nuclide $^{242}$Pu, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pu-242 (n,g|abs) sig|wid|ri|par

Figure A.25: Summary of the 82 EXFOR entries for the nuclide $^{235}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 3

U-235 (n,g|abs) sig|wid|ri|par
Figure A.26: Summary of the 82 EXFOR entries for the nuclide $^{235}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 3.

U-235 (n,g|abs) sig|wid|ri|par

Figure A.27: Summary of the 82 EXFOR entries for the nuclide $^{235}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 3 of 3.

U-235 (n,g|abs) sig|wid|ri|par
Figure A.28: Summary of the 200 EXFOR entries for the nuclide $^{238}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 5

Figure A.29: Summary of the 200 EXFOR entries for the nuclide $^{238}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 2 of 5
Figure A.30: Summary of the 200 EXFOR entries for the nuclide $^{238}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 3 of 5

Figure A.31: Summary of the 200 EXFOR entries for the nuclide $^{238}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 4 of 5
Figure A.32: Summary of the 200 EXFOR entries for the nuclide $^{238}$U, and reactions (n,g), (n,abs), for entries matching the quantities sig, wid, ri, par; part 5 of 5

Figure A.33: Summary of the 100 EXFOR entries for the nuclide $^{241}$Am, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 3
Figure A.34: Summary of the 100 EXFOR entries for the nuclide $^{241}$Am, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 3

Am-241 (n,f) sig|wid|ri|par

Figure A.35: Summary of the 100 EXFOR entries for the nuclide $^{241}$Am, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 3 of 3

Am-241 (n,f) sig|wid|ri|par
Figure A.36: Summary of the 67 EXFOR entries for the nuclide $^{243}$Am, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Figure A.37: Summary of the 67 EXFOR entries for the nuclide $^{243}$Am, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 2
Figure A.38: Summary of the 10 EXFOR entries for the nuclide $^{242}$Cm, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Cm-242 (n,f) sig|wid|ri|par

Figure A.39: Summary of the 55 EXFOR entries for the nuclide $^{243}$Cm, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Cm-243 (n,f) sig|wid|ri|par
Figure A.40: Summary of the 55 EXFOR entries for the nuclide $^{243}$Cm, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Cm-243 (n,f) sig|wid|ri|par

Figure A.41: Summary of the 35 EXFOR entries for the nuclide $^{244}$Cm, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Cm-244 (n,f) sig|wid|ri|par
Figure A.42: Summary of the 52 EXFOR entries for the nuclide $^{245}$Cm, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Cm-245 (n,f) sig|wid|ri|par

Figure A.43: Summary of the 52 EXFOR entries for the nuclide $^{245}$Cm, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Cm-245 (n,f) sig|wid|ri|par
Figure A.44: Summary of the 24 EXFOR entries for the nuclide $^{246}$Cm, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 1.

Cm-246 (n,f) sig|wid|ri|par

Figure A.45: Summary of the 147 EXFOR entries for the nuclide $^{237}$Np, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 4.

Np-237 (n,f) sig|wid|ri|par
Figure A.46: Summary of the 147 EXFOR entries for the nuclide $^{237}$Np, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 4

Figure A.47: Summary of the 147 EXFOR entries for the nuclide $^{237}$Np, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 3 of 4
Figure A.48: Summary of the 147 EXFOR entries for the nuclide $^{237}$Np, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 4 of 4

Np-237 (n,f) sig|wid|ri|par

Figure A.49: Summary of the 49 EXFOR entries for the nuclide $^{238}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Pu-238 (n,f) sig|wid|ri|par
Figure A.50: Summary of the 49 EXFOR entries for the nuclide $^{238}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Pu-238 (n,f) sig|wid|ri|par

Figure A.51: Summary of the 73 EXFOR entries for the nuclide $^{240}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Pu-240 (n,f) sig|wid|ri|par
Figure A.52: Summary of the 73 EXFOR entries for the nuclide $^{240}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Pu-240 (n,f) sig|wid|ri|par

Figure A.53: Summary of the 118 EXFOR entries for the nuclide $^{241}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 3

Pu-241 (n,f) sig|wid|ri|par
Figure A.54: Summary of the 118 EXFOR entries for the nuclide $^{241}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 3

Pu-241 (n,f) sig|wid|ri|par

Figure A.55: Summary of the 118 EXFOR entries for the nuclide $^{241}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 3 of 3

Pu-241 (n,f) sig|wid|ri|par
Figure A.56: Summary of the 49 EXFOR entries for the nuclide $^{242}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Figure A.57: Summary of the 49 EXFOR entries for the nuclide $^{242}$Pu, and reactions (n,f), for entries matching the quantities sig, wid, ri, par; part 2 of 2
Figure A.58: Summary of the 61 EXFOR entries for the nuclide $^{12}$C, $^0$C, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Figure A.59: Summary of the 61 EXFOR entries for the nuclide $^{12}$C, $^0$C, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 2 of 2
Figure A.60: Summary of the 24 EXFOR entries for the nuclide $^{52}$Cr, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Cr-52 (n,el) sig|wid|ri|par

Figure A.61: Summary of the 62 EXFOR entries for the nuclide $^{56}$Fe, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 1 of 2

Fe-56 (n,el) sig|wid|ri|par
Figure A.62: Summary of the 62 EXFOR entries for the nuclide $^{56}$Fe, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Figure A.63: Summary of the 2 EXFOR entries for the nuclide $^{15}$N, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 1 of 1.
Figure A.64: Summary of the 27 EXFOR entries for the nuclide $^{16}$O, $^{18}$O, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Figure A.65: Summary of the 10 EXFOR entries for the nuclide $^{208}$Pb, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 1 of 1
Figure A.66: Summary of the 58 EXFOR entries for the nuclide $^{238}$U, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 1 of 2

U-238 (n,el) sig|wid|ri|par

Figure A.67: Summary of the 58 EXFOR entries for the nuclide $^{238}$U, and reactions (n,el), for entries matching the quantities sig, wid, ri, par; part 2 of 2

U-238 (n,el) sig|wid|ri|par
Figure A.68: Summary of the 2 EXFOR entries for the nuclide $^{241}$Am, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Am-241 (n,inl) sig|wid|ri|par

Figure A.69: Summary of the 4 EXFOR entries for the nuclide $^{243}$Am, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Am-243 (n,inl) sig|wid|ri|par
Figure A.70: Summary of the 36 EXFOR entries for the nuclide $^{209}$Bi, and reactions $\text{n,inl}$, for entries matching the quantities sig, wid, ri, par; part 1 of 1

Figure A.71: Summary of the 50 EXFOR entries for the nuclide $^{12}$C, $^{13}$C, and reactions $\text{n,inl}$, for entries matching the quantities sig, wid, ri, par; part 1 of 2
Figure A.72: Summary of the 50 EXFOR entries for the nuclide $^{12}$C, $^{13}$C, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Figure A.73: Summary of the 109 EXFOR entries for the nuclide $^{56}$Fe, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 3
Figure A.74: Summary of the 109 EXFOR entries for the nuclide $^{56}$Fe, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 2 of 3

Figure A.75: Summary of the 109 EXFOR entries for the nuclide $^{56}$Fe, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 3 of 3
Figure A.76: Summary of the 37 EXFOR entries for the nuclide $^{23}$Na, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Na-23 (n,inl) sig|wid|ri|par

Figure A.77: Summary of the 19 EXFOR entries for the nuclide $^{0}$Pb, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pb-0 (n,inl) sig|wid|ri|par
Figure A.78: Summary of the 16 EXFOR entries for the nuclide $^{206}$Pb, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pb-206 (n,inl) sig|wid|ri|par

Figure A.79: Summary of the 15 EXFOR entries for the nuclide $^{207}$Pb, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pb-207 (n,inl) sig|wid|ri|par
Figure A.80: Summary of the 17 EXFOR entries for the nuclide $^{239}\text{Pu}$, and reactions (n,\text{inl}), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pu-239 (n,\text{inl}) sig|wid|ri|par

Figure A.81: Summary of the 2 EXFOR entries for the nuclide $^{240}\text{Pu}$, and reactions (n,\text{inl}), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pu-240 (n,\text{inl}) sig|wid|ri|par
Figure A.82: Summary of the 1 EXFOR entries for the nuclide $^{242}$Pu, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Pu-242 (n,inl) sig|wid|ri|par

Figure A.83: Summary of the 37 EXFOR entries for the nuclide $^{28}$Si, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 1

Si-28 (n,inl) sig|wid|ri|par
Figure A.84: Summary of the 63 EXFOR entries for the nuclide $^{238}$U, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 1 of 2

U-238 (n,inl) sig|wid|ri|par

Figure A.85: Summary of the 63 EXFOR entries for the nuclide $^{238}$U, and reactions (n,inl), for entries matching the quantities sig, wid, ri, par; part 2 of 2

U-238 (n,inl) sig|wid|ri|par
Figure A.86: Summary of the 17 EXFOR entries for the nuclide $^{90}$Zr, and reactions $(n,\text{inl})$, for entries matching the quantities $\text{sig, wid, ri, par}$; part 1 of 1

$^{90}$Zr $(n,\text{inl}) \text{ sig|wid|ri|par}$

Figure A.87: Summary of the 53 EXFOR entries for the nuclide $^{16}$O, $^{17}$O, and reactions $(n,\text{sct})$, $(n,\text{el})$, $(n,\text{inl})$, for entries matching the quantities $\text{sig, wid, ri, par}$; part 1 of 2

$^{16}$O/$^{17}$O $(n,\text{sct}|\text{el}|\text{inl}) \text{ sig|wid|ri|par}$
Figure A.88: Summary of the 53 EXFOR entries for the nuclide $^{16}$O, $^{17}$O, and reactions (n,sct), (n,el), (n,inel), for entries matching the quantities sig, wid, ri, par; part 2 of 2

Figure A.89: Summary of the 131 EXFOR entries for the nuclide $^{238}$U, and reactions (n,sct), (n,el), (n,inel), for entries matching the quantities sig, wid, ri, par; part 1 of 4
Figure A.90: Summary of the 131 EXFOR entries for the nuclide $^{238}$U, and reactions (n,sct), (n,el), (n,inl), for entries matching the quantities sig, wid, ri, par; part 2 of 4

U-238 (n,sct|el|inl) sig|wid|ri|par

Figure A.91: Summary of the 131 EXFOR entries for the nuclide $^{238}$U, and reactions (n,sct), (n,el), (n,inl), for entries matching the quantities sig, wid, ri, par; part 3 of 4

U-238 (n,sct|el|inl) sig|wid|ri|par
Figure A.92: Summary of the 131 EXFOR entries for the nuclide $^{238}$U, and reactions (n,sct), (n,el),
(n,inl), for entries matching the quantities sig, wid, ri, par; part 4 of 4.

Figure A.93: Summary of the 117 EXFOR entries for the nuclide $^{10}$B, and reactions (n,a), (n,abs),
(n,tot), for entries matching the quantities sig, wid, ri, par; part 1 of 3.
Figure A.94: Summary of the 117 EXFOR entries for the nuclide $^{10}$B, and reactions (n,a), (n,abs), (n,tot), for entries matching the quantities sig, wid, ri, par; part 2 of 3

Figure A.95: Summary of the 117 EXFOR entries for the nuclide $^{10}$B, and reactions (n,a), (n,abs), (n,tot), for entries matching the quantities sig, wid, ri, par; part 3 of 3
Figure A.96: Summary of the 37 EXFOR entries for the nuclide $^{12}\text{C}$, $^{16}\text{O}$, and reactions (n,a), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

C-12|C-0 (n,a|abs) sig|wid|ri|par

Figure A.97: Summary of the 27 EXFOR entries for the nuclide $^{16}\text{O}$, $^{16}\text{O}$, and reactions (n,a), (n,abs), for entries matching the quantities sig, wid, ri, par; part 1 of 1

O-16|O-0 (n,a|abs) sig|wid|ri|par