# ENDF/B-VIII.0 NUCLEAR DATA SENSITIVITY

# AND UNCERTAINTY ANALYSIS OF KEY

# SAFETY-RELEVANT REACTIVITY COEFFICIENTS

# FOR THE ALFRED CORE

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**Abstract**

ENEA has a long-lasting expertise in the design of Gen IV nuclear reactors, in particular the ones cooled by liquid Lead (LFRs). In the EU context, through the participation to the FALCON Consortium, ENEA is pursuing all the activities required to support the construction of ALFRED – the European demonstrator of the LFRs – in Romania. S/U analyses are a paramount step for the licensing of such an innovative reactor. In fact, no previous LFR experience can be used for validating neutronic calculations justifying the design of the core, so that a thorough assessment of the calculation uncertainties must be used in front of the safety authorities asked to license ALFRED construction. Indeed, S/U analyses are used for establishing the adequateness of the assumed safety margins, as one of the key goals in designing the demonstrator, by verifying that such safety margins cope – with the aimed confidence – with the relative uncertainties. The objective of the paper is to present the S/U analysis of the ALFRED reactor in order to assess the impact of the nuclear data uncertainties on the core reactivity and on the most important safety-relevant reactivity effects: *e.g.*, coolant density effect, temperature-related effects, control rod worth, delayed neutron fraction, etc. Both the sensitivity and uncertainty analyses are here presented so to give the full picture of the parameters investigated, outlining what are the most important isotope-reaction couples both from the purely physical and nuclear data quality standpoints. S/U analyses are performed using one of the most up-to-date nuclear data evaluations, ENDF/B-VIII.0, prepared in a special format readable by the selected neutronic code, ERANOS, which does not accept libraries in the standard ENDF-6 format. Moreover, regarding the needed covariances, in order to avoid inconsistencies and with the aim of enhancing the confidence on the obtained results, a new homemade one also based on the state-of-the-art evaluation ENDF/B-VIII.0, was generated and used.

## INTRODUCTION

The development of new powerful computers and high-performance analytical tools, along with the reduction of the approximations due to new methods implemented in the algorithms for the resolution of the transport equation, pushed nuclear cross-sections data as the main source of uncertainty in neutronic analysis. Efforts in nuclear data improvements and therefore in advanced simulations are one of the pathways that could allow designers to set safety margins so to permit broader flexibility to optimization while complying with ambitious safety goals, as an enabling asset especially in designing innovative systems. In such systems, indeed, the lesser operational experience takes important arguments in support not only of design optimization, as required for the simultaneous achievement of enhanced performance and ambitious safety claims (typically associated to innovative reactor concepts), but also of the due justification of the proposed design in terms of meeting the assumed safety limits.

Sensitivity and Uncertainty (S/U) analyses are therefore a paramount step for the licensing of innovative Gen-IV nuclear systems, and particularly for the pioneering ones, such as ALFRED [1], appointed to the role of Lead-cooled Fast Reactor (LFR) technology demonstrator in Europe. In fact, no previous experience can be claimed to directly validate neutronic calculations justifying the design of the core, so that a thorough assessment of the involved uncertainties must be used in front of the safety authorities asked to license the ALFRED construction and operation. With the aim of strengthening the accuracy and predictive capabilities of the calculations for the ALFRED reactor, thus at establishing due confidence for the setting of the associated safety margins, a S/U analysis was performed for its main safety-relevant parameters, also as a propaedeutic step for a successive library adjustment [2] The effective multiplication factor ($k\_{eff}$), the coolant density effect, the fuel Doppler effect, the protection system worth and the effective delayed neutrons fraction ($β\_{eff}$) were evaluated, and the associated S/U results are presented in the paper outlining what are the most important isotope-reaction couples both from the purely physical and nuclear data quality standpoints.

It is worth mentioning that the work, initiated by the needs of the ALFRED project, also matches with the framework of the OECD Nuclear Energy Agency (NEA) Working Party on International Nuclear Data Evaluation and Cooperation (WPEC) Sub-Group 46 [3], aimed, among the other objectives, at launching a target accuracy requirements exercise pinpointing a priority list of isotopes, through S/U analyses for the most important key safety-relevant core parameters for several reference systems, including the ALFRED reactor.

## nUCLEAR DATA LIBRARY AND COVARIANCE MATRIX

The European Reactor Analysis Optimised System (ERANOS) [4] is one of the most advanced deterministic tools developed in Europe for neutronic analysis of fast reactors. It was successfully used in the past to perform neutron physics commissioning pre-tests for the Phénix and Superphénix reactors [5], and extensively used both for reactor physics and reactor design analysis for a wide range of innovative systems (*i.e.*, SFR, GFR, etc.). ERANOS is conceived as a very comprehensive suite, characterized by extreme flexibility and modularity which were exploited in the paper for performing S/U analyses for the main integral parameters of the ALFRED core. However, even the latest, publicly available release, ERANOS 2.3 [6] (hereafter labelled ERANOS) is still coupled with neutron cross-section libraries derived from the JEFF-3.1 and ENDF/B-VI.8 evaluated nuclear data files, both released a long a time ago. This is in contrast with the needs of state-of-the-art nuclear data evaluations for performing a sound and coherent S/U analysis, by leveraging on the latest worldwide experiments carried out in several facilities in the world and on the most accurate and detailed computational nuclear models.

Moreover, the ERANOS package is distributed without covariance data which must therefore be supplied externally. In fact, the JEFF-3.1 and ENDF/B-VI.8 evaluations, in contrast to the latest releases, do not systematically include nuclear data uncertainties for the major nuclides and reactions, requiring the latter to be taken from other sources and/or assessed via expert judgment: this is the case of the so-called BOLNA matrix [7]. The use of recent libraries along with a covariance matrix not generated by the same original set of evaluations would introduce inconsistencies between the nuclear data reference values and their uncertainties, ultimately reducing the confidence on the results of S/U analyses. Therefore, being a matter of facts that uncertainty overestimation is brought about by the use of dated covariances, moving to more recent evaluations opens up the possibility of more realistic estimates.

In this work, to enforce coherency, S/U evaluations were performed using a set of cross-sections and their relative covariance matrices both generated from the recent ENDF/B-VIII.0 evaluations by means of codes specifically developed for the purpose. Concerning cross-sections, the entire generation process was firstly tested with the old JEFF3.1 libraries with the aim of verifying its correctness via direct comparison of cell and core calculations performed with the official version of the same library released with the ERANOS code. Moreover, partial cross-sections were combined into five primary reactions named ELASTIC, CAPTURE, FISSION, INELASTIC, and NxN some of which include different MT numbers. In particular, CAPTURE includes processes in which no neutrons are emitted (MT=102 + 103 + …), INELASTIC includes any process emitting one neutron only and leaving the target nuclide in an excited state (MT=4 + 22 + …), and NxN includes any process emitting several neutrons, except fission.

Uncertainty-side, only MF=31 and MF=33 files containing, respectively, uncertainties information on fission neutron multiplicities and cross-sections, were processed, while covariances of the fission spectrum and in general of information on secondary particles energy (MF=35), angle distributions (MF=34) and delayed data were not used.

To perform a physically sound S/U analysis, it was also paramount to select a multigroup energy structure able to capture the physics of the system, so to increase confidence in the results reliability.

According to this aim, all the most common energy structures used in different specialized laboratories worldwide were analysed. The so called LANL 80 group structure [8, 9] emerged not only as a reasonable trade-off in the number of groups (detail of description vs. computational cost), but also with an appropriate energy distribution of the groups themselves, with an increased refinement over the most populated region of the ALFRED spectrum. Additionally, the structure almost fit exactly the 1968 standard group structure used in ERANOS, except in some of the very lowest energy intervals. To account for this, it was realized that the modification to such energy intervals so as to obtain a perfect match with the energy boundaries as used in ERANOS, minor in nature, would be physically justified by the very low neutronic importance of the involved intervals. For all the reasons above, the obtained energy group structure, named “LANL 80 mod”, was thus used for the entire work hereby. It is worth stressing that, since ERANOS does not accept libraries in ENDF-6 format [10], the one worldwide used for evaluated nuclear data files, both the library and the covariance matrix were processed to generate their analogous versions in the peculiar format required (named ECCOLIB) [11]. The new library and its relative covariance matrix, in the right ECCOLIB format, were then used for performing the S/U analysis for the main integral parameters of the ALFRED core.

## THE ERANOS MODEL of ALFRED AND ITS OPTIMIZATION

As already mentioned, ERANOS modules were used to perform a complete S/U analysis for the most up-to-date design of the ALFRED core layout [1]. ERANOS requires a solution of the transport equation with the finite difference and SN approximations in a 2D geometry model to effectively execute S/U analyses, which can be conveniently done using the dedicated tools available for the 2D BISTRO solver [12]. Given the highly symmetric nature of the ALFRED core, the reference 2D geometry was chosen as the cylindrical (RZ) one. In order to obtain an optimized 2D model, the ALFRED core was firstly modelled in ERANOS with the exact 3D hexagonal geometry, making a spatial discretization for all the assemblies in the core, and trying to identify for all the assembly types, regions with different neutronic features. Starting from the reference 3D hexagonal model, a cylindrical transformation was performed in order to obtain the 2D one. In this operation, the controls for neutronic equivalence went beyond the sole preservation of the overall volume of each region. Specifically, a thorough study was done to fix the choice of the radii of the interfaces between the fuel and control rods and safety devices. Since, at the basis of S/U analyses, stands the correct evaluation of both the direct and adjoint fluxes, this paramount criterion was used to fix the mentioned radii, so to preserve their shape relatively to the reference one calculated with the reference 3D hexagonal model by the TGV/VARIANT [13] solver. The best cylindrical configuration was thus found, bringing to a radial representation of the cylindrical model (see Fig.1) composed of 24 different regions, where the active core zones are modelled along with the radially and axially surrounding structures (*e.g.*, thermal insulators, gas plenum, diagrid, etc.). The multi-group constants representing the different core regions were obtained separately, with the European Cell Code ECCO [14] by adopting 2D heterogeneous geometry models.

Fig. 1. Cylindrical transformation of the ALFRED core model.

## S/U analysis of the safety-relevant parameters

Perturbation Theory (PT) has always played a fundamental role also in fields other than neutron physics [15, 16]: in the present paper it was used to perform S/U analyses for the most relevant key-safety effects for the ALFRED core. The analyses of fundamental physical parameters – such as the effective multiplication factor, the coolant density effect, the fuel Doppler effect, the protection system worth and the effective delayed neutrons fraction – were carried out for the ALFRED reactor using the PT, either in the so-called Standard Perturbation Theory (SPT) and in Generalized one (GPT), or in its Equivalent Generalized formulation (EGPT). For the effective multiplication factor, SPT was used since it is specifically tailored for the latter. GPT was instead used for the effective delayed neutron fraction, leveraging on its formulation which is specifically applicable to bilinear ratios of flux and importance. Finally, EGPT was used for all the reactivity effects since it has the ability to considerably simplify the original problem to a difference between two SPT calculations for each of the two reactivity states.

### The effective multiplication factor

The effective neutron multiplication factor was the first parameter investigated using SPT methods, since it integrates major aspects of the core neutronics. As shown in Table 1, from the sensitivity side, the isotope-reaction couples which influence $k\_{eff}$ the most are all related to the fission event except for:

1. the capture of 238U and 239Pu, the two most abundant isotopes in the fuel;
2. and to a minor degree, the scattering, inelastic and elastic respectively, of 238U and 16O, which affect the spectrum hardness.

TABLE 1. $k\_{eff}$ S/U DUE TO NUCLEAR DATA MAJOR ISOTOPE-REACTION COUPLES

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | **Sensitivity** |  |  | **Uncertainty** |  |
| **Isotope** | **Reaction** | **Value** | **Isotope** | **Reaction** | **Value** |
| 239Pu | $$ν$$ | 6.90∙10-1 | 239Pu | fission | 5.70∙10-3 |
| 239Pu | fission | 4.88∙10-1 | 239Pu | capture | 2.70∙10-3 |
| 238U | capture | -1.75∙10-1 | 238U | capture | 2.41∙10-3 |
| 241Pu | $$ν$$ | 1.03∙10-1 | 239Pu | $$ν$$ | 1.88∙10-3 |
| 238U | $$ν$$ | 8.53∙10-2 | 238U | inelastic | 1.39∙10-3 |
| 241Pu | fission | 7.42∙10-2 | 238U | $$ν$$ | 1.07∙10-3 |
| 240Pu | $$ν$$ | 7.17∙10-2 | 207Pb | inelastic | 8.44∙10-4 |
| 239Pu | capture | -5.18∙10-2 | 239Pu | inelastic | 8.34∙10-4 |
| 238U | fission | 5.06∙10-2 | 240Pu | capture | 8.29∙10-4 |
| 240Pu | fission | 4.85∙10-2 | 56Fe | capture | 8.03∙10-4 |
| 238U | inelastic | -3.76∙10-2 | 208Pb | elastic | 7.76∙10-4 |
| 16O | elastic | -3.53∙10-2 | 206Pb | inelastic | 7.20∙10-4 |
| **Total** |  | 1.31∙100 | **Total** |  | 7.68∙10-3 |

Standing on the sensitivities above, the total uncertainty was found to be 768 pcm, well below what predicted with legacy covariances (*i.e.*, 1186 pcm) [17]. At the level of single isotope contributions, the major one is still related to the fission of 239Pu. Significant contributions come from also from the scattering and capture for 239Pu and 238U nuclides, as well as from the number of neutrons ($ν$) generated by 239Pu fission, stressing the importance of reducing uncertainties on such reactions.

### Fuel Doppler Coefficient

The fuel Doppler coefficient is a fundamental safety-related parameter, especially in overpower-type transients. For this effect, an analysis was performed via EGPT methods for a temperature change from 1200 K up to 2073 K, roughly corresponding to the fuel temperature increase in the enveloping unprotected transient of overpower used for the ALFRED safety assessment [18]. The reference value of the Doppler coefficient has been found to be -0.415 pcm K-1. From the sensitivity side, as visible from Table 2, the Doppler coefficient is dominated by 239Pu production (including fission and $ν$), the 16O inelastic scattering and the 56Fe capture. Regarding uncertainty, the total value has been found to be 2.94∙10-2 and the main contributions come from scattering and capture but for the leading one, coming again from 239Pu fission.

TABLE 2. FUEL DOPPLER REACTIVITY EFFECT S/U DUE TO NUCLEAR DATA FOR MAJOR ISOTOPE-REACTION COUPLES

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | **Sensitivity** |  |  | **Uncertainty** |  |
| **Isotope** | **Reaction** | **Value** | **Isotope** | **Reaction** | **Value** |
| 239Pu | fission | -1.07∙100 | 239Pu | fission | 1.71∙10-2 |
| 16O | elastic | 7.58∙10-1 | 239Pu | capture | 1.67∙10-2 |
| 239Pu | $$ν$$ | -6.29∙10-1 | 238U | inelastic | 9.77∙10-3 |
| 56Fe | capture | -3.36∙10-1 | 239Pu | inelastic | 6.62∙10-3 |
| 238U | elastic | 1.89∙10-1 | 56Fe | inelastic | 4.84∙10-3 |
| 238U | inelastic | 1.83∙10-1 | 238U | inelastic | 4.20∙10-3 |
| 238U | $$ν$$ | -1.75∙10-1 | 238U | capture | 3.25∙10-3 |
| 238U | capture | 1.51∙10-1 | 238U | capture | 3.16∙10-3 |
| 240Pu | $$ν$$ | -1.44∙10-1 | 240Pu | inelastic | 2.74∙10-3 |
| 238U | fission | -1.35∙10-1 | 238U | inelastic | 2.68∙10-3 |
| 240Pu | fission | -1.29∙10-1 | 240Pu | elastic | 2.58∙10-3 |
| 240Pu | capture | -1.07∙10-1 | 240Pu | capture | 2.53∙10-3 |
| **Total** |  | -1.27∙100 | **Total** |  | 2.94∙10-2 |

### Coolant density effect

When a change in the coolant temperature occurs, the resulting density variation determines a reactivity change due to three simultaneous and concurrent phenomena: the increase in neutron leakage, the hardening of the spectrum and the reduction in capture. In the work, a 20% reduction of the coolant density was considered in the whole active region for ensuring a non-negligible reactivity variation, so avoiding numerical problems in the EGPT process, while assuring that the linear regime of this effect is not exceeded. The reference value of the coolant density coefficient was found to be +0.07 pcm K-1. From the sensitivity results in Table 3, the concomitant effects previously introduced are clearly visible, with the ranking of the cross-sections changed (with respect to nominal conditions as per Table 1) according to the magnification of such effects: fission-related cross-sections, including for even nuclides, because of spectrum hardening, scattering- and capture-related cross-sections because of reduced coolant captures and higher neutron leakage. Similarly to what was observed for the $k\_{eff}$ and fuel Doppler effect, the 239Pu fission yields both the highest sensitivity and uncertainty values. Moreover, uncertainties are dominated by capture and scattering, with lead isotopes among the major contributors. In particular, significant effects are due to 206Pb and 207Pb inelastic reactions highlighting the importance in reducing the associated uncertainty. The total contribution was found to be significant, 1.47∙10-1, albeit not such to determine a marked impact even to the most demanding design extension conditions [19].

TABLE 3. COOLANT DENSITY REACTIVITY EFFECT DUE TO NUCLEAR DATA FOR MAJOR ISOTOPE-REACTION COUPLES

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | **Sensitivity** |  |  | **Uncertainty** |  |
| **Isotope** | **Reaction** | **Value** | **Isotope** | **Reaction** | **Value** |
| 239Pu | $$ν$$ | -2.93∙100 | 239Pu | fission | 8.64∙10-2 |
| 239Pu | fission | -2.24∙100 | 206Pb | capture | 7.39∙10-2 |
| 238U | capture | 2.14∙100 | 239Pu | inelastic | 3.58∙10-2 |
| 238U | $$ν$$ | 2.00∙100 | 208Pb | inelastic | 3.50∙10-2 |
| 238U | fission | 1.16∙100 | 238U | inelastic | 2.97∙10-2 |
| 16O | elastic | 1.00∙100 | 239Pu | inelastic | 2.96∙10-2 |
| 206Pb | inelastic | 9.35∙10-1 | 207Pb | capture | 2.94∙10-2 |
| 241Pu | $$ν$$ | -6.82∙10-1 | 208Pb | capture | 2.80∙10-2 |
| 207Pb | inelastic | -6.63∙10-1 | 238U | inelastic | 2.38∙10-2 |
| 238U | inelastic | -6.08∙10-1 | 204Pb | inelastic | 2.22∙10-2 |
| 239Pu | capture | -5.69∙10-1 | 56Fe | elastic | 1.69∙10-2 |
| 240Pu | $$ν$$ | -5.38∙10-1 | 206Pb | capture | 1.44∙10-2 |
| **Total** |  | 4.11∙100 | **Total** |  | 1.47∙10-1 |

### Effective delayed neutron fraction

The effective delayed neutron fraction ($β\_{eff}$) is a fundamental parameter to accurately know, because of its central role for reactor control.

A GPT approach is followed in the paper to derive sensitivities, but since ERANOS does not implement modules for directly performing such a calculation, the flexibility offered by its metalanguage “LU” was exploited writing an ad-hoc procedure to set up and solve the resulting equations. Specifically, the functional can be built starting from the $β\_{eff}$ definition as:

|  |  |
| --- | --- |
| $^\_{eff}=\sum\_{n=1}^{NP}^\_{eff,n} with ^\_{eff,n}= \frac{\left〈Φ^{+}χ\_{n}^{d}, β\_{n}νΣ\_{f}Φ\right〉}{\left〈Φ^{+}, FΦ\right〉}$. | (1) |

By imposing the critical core constraints, the Lagrange multipliers equations are found as:

|  |  |
| --- | --- |
| $\left(A^{+}-\frac{1}{k\_{eff}}F^{+}\right)\tilde{ψ}\_{n}^{+}=\frac{β\_{n}\left[Φ^{+}χ\_{n}^{d}\right]νΣ\_{f}\left(r,E\right)}{\left〈χ\_{n}^{d}Φ^{+},β\_{n}νΣ\_{f}Φ\right〉}- \frac{\left[Φ^{+}χ\right]νΣ\_{f}\left(r,E\right)}{\left〈Φ^{+}, FΦ\right〉} with \tilde{ψ}\_{n}^{+}=\frac{ψ^{+}}{^\_{eff,n}}$, | (2) |
| $\left(A-\frac{1}{k\_{eff}}F\right)\tilde{ψ}\_{n}= \frac{\left[β\_{n}νΣ\_{f}Φ\right]χ\_{n}^{d}\left(E\right)}{\left〈χ\_{n}^{d}Φ^{+},β\_{n}νΣ\_{f}Φ \right〉}- \frac{\left[νΣ\_{f}Φ\right]χ\left(E\right)}{\left〈Φ^{+}, FΦ\right〉} with \tilde{ψ}\_{n}=\frac{ψ}{^\_{eff,n}}$, | (3) |

where $NP$ is the number of precursors families, $β\_{n}$ is the delayed neutrons fraction of the $n$-th family, $χ$ is the fission spectrum, $χ\_{n}^{d}$ is the emission spectrum of the delayed neutrons of family $n$ and […] indicates integration over energy. Equations (2) and (3) must be solved for each of the $NP$ precursors families.

The sensitivity coefficients can be finally derived as:

|  |  |
| --- | --- |
| $S\_{^\_{eff,n}}= \frac{σ}{^\_{eff,n}} \frac{d^\_{eff,n}}{dσ}= \frac{σ}{^\_{eff,n}} \left\{\frac{∂^\_{eff,n}}{∂σ}- \left〈ψ^{+},\left(\frac{∂A}{∂σ}- \frac{1}{k\_{eff}} \frac{∂F}{∂σ}\right)Φ\right〉- \left〈ψ , \left(\frac{∂A^{+}}{∂σ}- \frac{1}{k\_{eff}}\frac{∂F^{+}}{∂σ}\right)Φ^{+}\right〉\right\} $, | (4) |

with

|  |  |
| --- | --- |
| $\frac{σ}{^\_{eff,n}} \frac{∂^\_{eff,n}}{∂σ}= \frac{\left(\left[χ\_{n}^{d} Φ^{+}\right]β\_{n}υΣ\_{f}Φ\right)\_{i,g,d}}{\left〈χ\_{n}^{d}Φ^{+},β\_{n}υΣ\_{f}Φ \right〉}- \frac{\left(\left[χΦ^{+}\right]υΣ\_{f}Φ\right)\_{i,g,d}}{\left〈Φ^{+}, FΦ\right〉}$, | (5) |

and then:

|  |  |
| --- | --- |
| $S\_{^\_{eff}}= \frac{σ}{^\_{eff}} \frac{d^\_{eff}}{dσ}= \sum\_{n=1}^{NP}\frac{^\_{eff,n}}{^\_{eff}} S\_{^\_{eff,n}} $, | (6) |

From solving the GPT problem with eight precursor families, a $β\_{eff}$ value of 345.9 pcm was found. S/U results are instead summarized in Table 4.

TABLE 4. COOLANT DENSITY REACTIVITY EFFECT S/U DUE TO NUCLEAR DATA FOR MAJOR ISOTOPE-REACTION COUPLES

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | **Sensitivity** |  |  | **Uncertainty** |  |
| **Isotope** | **Reaction** | **Value** | **Isotope** | **Reaction** | **Value** |
| 239Pu | fission | -4.29∙10-1 | 239Pu | fission | 4.89∙10-3 |
| 239Pu | $$ν$$ | -3.95∙10-1 | 238U | $$ν$$ | 4.76∙10-3 |
| 238U | $$ν$$ | 3.70∙10-1 | 238U | fission | 3.84∙10-3 |
| 238U | fission | 3.43∙10-1 | 239Pu | capture | 2.27∙10-3 |
| 238U | inelastic | 3.22∙10-2 | 238U | capture | 1.45∙10-3 |
| 56Fe | inelastic | 2.00∙10-2 | 239Pu | $$ν$$ | 1.32∙10-3 |
| 242Pu | $$ν$$ | 1.61∙10-2 | 207Pb | inelastic | 8.19∙10-4 |
| 240Pu | $$ν$$ | 1.60∙10-2 | 56Fe | inelastic | 5.16∙10-4 |
| 206Pb | inelastic | 1.57∙10-2 | 239Pu | inelastic | 4.86∙10-4 |
| 242Pu | fission | 1.26∙10-2 | 240Pu | fission | 4.21∙10-4 |
| 238Pu | fission | -1.10∙10-2 | 206Pb | inelastic | 3.87∙10-4 |
| 241Pu | $$ν$$ | 1.07∙10-2 | 207Pb | elastic | 3.62∙10-4 |
| **Total** |  | 8.66∙10-2 | **Total** |  | 8.40∙10-3 |

Sensitivity-wise, it is interesting to note that, despite not contributing to the overall sensitivity due to compensating effects between isotopes, $S\_{^\_{eff}} $ is dominated by the direct effect (relatively to the indirect one), coming from the fission and $ν$ of 239Pu and 238U. For this fundamental parameter, the total uncertainty was found to be 8.40∙10-3, so satisfactory low. Regarding single contributions, scattering- and capture-related reactions dominate, together with fission mainly via 239Pu and 238U.

### Control rod worth

The evaluation of the CRs worth was performed, via EGPT, moving them from a completely extracted to a completely inserted position so to simulate their intervention by scram during a generic accident sequence. The reference value of the CRs worth was found to be 9563 pcm. From the sensitivity-side, as shown in Table 5, the importance of fission (meaning both fission and $ν$) of 239Pu and 238U can be seen. These contributions, as expected, are negative, being fission a concurrent effect of the CRs absorption. The main positive contribution obviously comes from the 10B capture, the absorber material the CRs are made of. To what concerns uncertainties, the total contribution is quite low, 1.08∙10-2, driven by the low values of the uncertainties associated to the reactions top ranking in sensitivity: along with those for 239Pu and 238U, previously discussed, the low uncertainty contribution observed from 10B capture suggests that also this reaction is already known with a satisfactory degree of accuracy. The scattering term of 208Pb is instead one of the top-ranking contributors, though it was not so for the sensitivities (6th position), again stressing the importance of reducing uncertainties on such reactions involving fuel and coolant.

TABLE 5. CONTROL RODS WORTH S/U DUE TO NUCLEAR DAATA FOR THE MAJOR ISOTOPE-REACTION COUPLES

|  |  |  |  |  |  |
| --- | --- | --- | --- | --- | --- |
|  | **Sensitivity** |  |  | **Uncertainty** |  |
| **Isotope** | **Reaction** | **Value** | **Isotope** | **Reaction** | **Value** |
| 239Pu | fission | -6.09∙10-1 | 239Pu | fission | 8.22∙10-3 |
| 239Pu | $$ν$$ | -5.65∙10-1 | 208Pb | elastic | 2.81∙10-3 |
| 10B | capture | 2.76∙10-1 | 238U | $$ν$$ | 2.59∙10-3 |
| 238U | $$ν$$ | -2.07∙10-1 | 239Pu | capture | 2.31∙10-3 |
| 238U | fission | -1.59∙10-1 | 238U | capture | 1.93∙10-3 |
| 208Pb | elastic | -1.04∙10-1 | 206Pb | inelastic | 1.85∙10-3 |
| 240Pu | $$ν$$ | -9.85∙10-2 | 206Pb | elastic | 1.78∙10-3 |
| 240Pu | fission | -9.19∙10-2 | 207Pb | elastic | 1.65∙10-3 |
| 16° | elastic | -9.04∙10-2 | 239Pu | $$ν$$ | 1.58∙10-3 |
| 238U | elastic | -7.48∙10-2 | 56Fe | inelastic | 1.52∙10-3 |
| 238U | capture | 7.41∙10-2 | 10B | capture | 1.36∙10-3 |
| 241Pu | fission | -7.22∙10-2 | 207Pb | inelastic | 1.29∙10-4 |
| **Total** |  | -2.01∙100 | **Total** |  | 1.08∙10-2 |

## CONCLUSION

Due to the demonstrator role of the ALFRED reactor, the verification that the assumed safety margins adequately take into account all the possible sources of uncertainty is a fundamental task. Applying this requirement in the neutronics field, an evaluation of all the uncertainties affecting each key reactivity effect governing the reactor dynamics also during an accident is fundamental in designing new reactor concepts. For this reason, a S/U analysis for ALFRED was performed using the most updated ENDF/B-VIII.0 nuclear data library, along with covariances generated from the latter using dedicated processes.

From the S/U analyses of the key integral parameters of the ALFRED core, including the effective multiplication factor, the fuel Doppler coefficient, the coolant density coefficient, the CRs worth and the effective delayed neutrons fraction, several important aspects have emerged:

* fission-related cross-sections (meaning both fission and $ν$) are among the most important contributors from a sensitivity standpoint, while scattering-related cross-sections, especially for lead isotopes, and capture reactions, especially for 238U and 239Pu, are typically dominating the parameters’ uncertainties.
* the most notable exception is the 239Pu fission cross-section, which resulted to be the major contributor to the uncertainty of all safety-related parameters, due to its far dominating sensitivity.

The quantification of the uncertainties on the integral parameters allowed to verify the adequateness of the safety margins assumed in the design of ALFRED. This conclusion was easy to affirm, also thanks to the fact that, at the time the ALFRED core was set, the early analyses were performed with outdated data libraries, thus with much larger uncertainties. Indeed, by direct comparison with an S/U analysis performed with the JEFF3.1 library and the BOLNA covariance matrix on the same safety-relevant effects considered in this work it was clearly shown that the use of more recent datasets brings about a reduction of nuclear-data-related uncertainties.

Notwithstanding, the above could be used to derive information to effectively reduce uncertainties, to the benefit of follow-on LFR units to rely on safety-margins effectively narrowed down, while ensuring a very high level of protection and thereby offering cost-free room for optimizing their design. The most impacting need relates to increase accuracy in the 239Pu fission cross-section and in scattering and capture cross-sections of fuel and coolant isotopes, so to obtain the most relevant reduction of the necessary safety margins in future LFRs. To fully substantiate this “semi-quantitative” claim, detailed target accuracy assessments can be performed so to better guide new experimental activities or library adjustment processes to the exact energy ranges and with the refence objective values.

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