Quantification of Nuclear Data Uncertainty Propagation for Kinetic Parameter in Fast Reactors Using Different Covariance Libraries

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Abstract

The closed fuel cycle with full actinide recycling in fast reactors represents a crucial approach for promoting the sustainable development of fission energy. Full actinide fuel significantly impacts core kinetic parameters, and its high-fidelity simulation is severely constrained by nuclear data uncertainties, resulting in insufficient confidence in the prediction of key parameters. This issue has emerged as a central challenge in enhancing reactor safety performance and optimizing system design. The effective delayed neutron fraction (βeff) constitutes a crucial parameter in reactor kinetics that governs reactor controllability during transients. Conducting sensitivity and uncertainty analyses for this parameter enables scientific quantification of system safety margins, facilitates identification of the primary sources of uncertainty, and guides the refinement of nuclear data. Therefore, this study focuses on the sensitivity and uncertainty of βeff in fast reactors. Firstly, nuclear data covariance libraries were generated based on the latest evaluated nuclear data libraries ENDF/B-VIII.1, JENDL-5.0, and JEFF-3.3. Subsequently, a sensitivity and uncertainty analysis code for βeff was independently developed using perturbation theory. Finally, systematic sensitivity and uncertainty analyses were performed on representative fast reactors such as ABTR, MET-1000, and CiADS. The calculation results indicate significant differences in the uncertainty of βeff obtained from different covariance libraries, with values ranging from 1% to 4%. Compared to the results from JENDL-5.0, the covariance libraries of ENDF/B-VIII.1 and JEFF-3.3 tend to significantly underestimate the uncertainty of βeff. This uncertainty primarily originates from nuclear parameters such as the number of fission neutrons (including both prompt and delayed neutrons), fission cross-section, and fission spectrum, with the main contributing energy range concentrated in the keV to MeV region.

Full Text

Preamble

Quantification of Nuclear Data Uncertainty Propagation for Kinetic Parameters in Fast Reactors Using Different Covariance Libraries

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The closed fuel cycle with full actinide recycling in fast reactors represents a crucial approach for promoting the sustainable development of fission energy. Full actinide fuel significantly affects core kinetic parameters, and its high-fidelity simulation is severely constrained by nuclear data uncertainties, leading to insufficient confidence in the prediction of key parameters. This issue has become a central challenge in enhancing reactor safety performance and optimizing system design. The effective delayed neutron fraction (β{eff}) *is a crucial parameter in reactor kinetics that governs reactor controllability during transients. Conducting sensitivity and uncertainty analyses on* β{eff} enables scientific quantification of system safety margins, helps pinpoint the main sources of uncertainty, and guides the refinement of nuclear data.

Therefore, this study focuses on the sensitivity and uncertainty of β{eff} *in fast reactors. First, nuclear data covariance libraries were generated based on the latest evaluated nuclear data libraries ENDF/B-VIII.1, JENDL-5.0, and JEFF-3.3. Subsequently, a sensitivity and uncertainty analysis code for* β{eff} was independently developed using perturbation theory. Finally, systematic sensitivity and uncertainty analyses were performed on representative fast reactors such as ABTR, MET-1000, and CiADS. The calculation results indicate that there are significant differences in the uncertainty of β{eff} *obtained from different covariance libraries, with values ranging from 1% to 4%. Compared to the results from JENDL-5.0, the covariance libraries of ENDF/B-VIII.1 and JEFF-3.3 tend to significantly underestimate the uncertainty of* β{eff}. Its uncertainty primarily arises from nuclear parameters such as the number of fission neutrons (including both prompt and delayed neutrons), fission cross-section, and fission spectrum, with the main contributing energy range concentrated in the keV to MeV region.

Keywords: Fast reactors; Sensitivity and uncertainty analysis; Covariances of nuclear data; Kinetic Parameter

Introduction

To address the challenges of uranium resource depletion and the accumulation of long-lived radioactive waste, six Generation IV nuclear energy system concepts have been proposed internationally. Among these, five are fast reactor systems, underscoring their dominant role in advanced nuclear technology [1]. Owing to advantages such as a hard energy spectrum and a high number of effective fission neutrons, fast reactors facilitate fuel breeding and transmutation of radioactive waste [2]. When integrated with closed fuel cycle technology, fast reactors can increase the utilization rate of natural uranium from less than 1% in thermal reactors to over 60% [3], establishing fast reactors as a critical pathway toward sustainable fission nuclear energy. Loading of all-actinide fuel containing uranium, plutonium, and minor actinides will significantly affect kinetic parameters of the reactor, especially the effective delayed neutron fraction (*β{eff}). Minor actinides (MAs) such as neptunium, americium, and curium generally exhibit lower* β{eff} values compared to those of uranium [4]. Since β_{eff} defines the upper limit of supercritical reactivity in a critical reactor, a reduction in its value raises challenges for operational safety. To avoid exceeding safety design criteria, the maximum permissible loading of MAs in critical reactors is typically restricted [5]. Thus, accurate determination of kinetic parameters is essential to ensure the safe operation and control of advanced nuclear energy systems.

In the design and safety analysis of early nuclear reactor systems, methodologies based on conservative assumptions and input conditions were typically adopted to ensure safety by incorporating substantial safety margins [6]. However, this overly conservative approach not only significantly compromised the economic efficiency of nuclear reactor systems but also revealed limitations in practical safety assurance. Following the Fukushima nuclear accident, the international community has raised stricter safety requirements for nuclear reactors [7], requiring confidence levels for numerical results based on high-precision modeling and simulation. This has propelled "Best Estimate plus Uncertainty Analysis (BEPU)"to become a research hotspot in reactor numerical simulation [8–13]. By integrating advanced computational models with systematic uncertainty quantification, this methodology aims to balance safety and economic performance while providing scientifically rigorous support for reactor design optimization and risk management [14]. The sources of uncertainty in nuclear reactor numerical simulations are primarily divided into two categories [15, 16]: engineering and computational. The former involves practical engineering deviations such as fuel manufacturing tolerances and measurement errors in power plant parameters, while the latter stems from errors in input parameters, model simplifications and approximations, and numerical discretization methods. These uncertainties propagate and accumulate through the modeling of physical processes, directly affecting the assessment of system safety and economic performance. In recent years, with the development of nuclear reactor physics calculation methods, the levels of uncertainty introduced by mathematical-physical model simplifications and numerical discretization have been significantly reduced [17, 18]. Currently, uncertainties in input parameters, particularly nuclear data, have become the most critical source of uncertainty in reactor physics calculations [19, 20]. To quantify the impact of nuclear data uncertainties on core parameters, sensitivity and uncertainty (S/U) analysis techniques have been developed [21–26]. These techniques systematically evaluate the contribution of nuclear data to computational outputs and identify the dominant sources of uncertainty, thereby providing a scientific basis for prioritizing the order of experimental measurements and theoretical evaluations for nuclear data.

Current research on nuclear data uncertainty quantification primarily focuses on parameters such as the effective multiplication factor (k_{eff}) and power distribution [27–30]. Less attention has been paid to the uncertainty quantification of kinetic parameters like the effective delayed neutron fraction *(β{eff}), primarily due to the lack of nuclear data for* β{eff} uncertainty analysis in early nuclear data evaluation libraries [31]. For instance, the ENDF/B-VII.1 library does not include uncertainty data for average numbers of delayed neutrons (_d) of ^{235}U and ^{238}U [32]; the JEFF-3.2 library is supplied with the _d uncertainties of ^{233}U and ^{241}Pu; the CENDL-3.2 library lacks such uncertainty information entirely [33]; and the BROND-3.1 library also omits _d uncertainty data for ^{233}U. Even today, significant variations persist in the completeness of delayed neutron uncertainty data across different evaluation libraries.

Bostelmann et al. quantified the uncertainty of *β{eff} for the MET-1000 fast reactor as 1.55% using covariance data from the ENDF/B-VII.1 library [34]; Kodeli et al. performed uncertainty analysis on* β{eff} for the accelerator-driven lead-bismuth cooled fast reactor MYRRHA based on the ENDF/B-VII.1 and JEFF-3.3 libraries, obtaining uncertainties of 0.79% and 1.10%, respectively [35]; Castelluccio et al. calculated the *β{eff} uncertainty of 0.84% for the ALFRED lead-cooled fast reactor using the ENDF/B-VIII.0 library [36]. However, the absence of complete _d uncertainty information for the all-actinide fuel in the nuclear data libraries used by these studies may lead to an underestimation of* β{eff} uncertainty.

This paper presents an approach for quantifying the propagation of nuclear data uncertainties in calculations of *β{eff}. The approach encompasses the development of a nuclear data covariance library, sensitivity coefficient analysis, uncertainty quantification, and the identification of key uncertainty sources. Based on the latest nuclear data evaluations [37]—ENDF/B-VIII.1, JENDL-5.0, and JEFF-3.3—three 33-group covariance libraries were generated using the nuclear data processing code NJOY2016 [38]. These covariance libraries include covariance information for the nuclear reaction cross-sections, average numbers of delayed neutrons, average numbers of prompt neutrons, fission neutron spectra, and other relevant nuclear data. The sensitivity profiles of* β{eff} with respect to nuclear data were computed via adjoint-weighted perturbation theory. These coefficients were subsequently combined with multi-group nuclear data covariance matrices to assess the uncertainty propagation. To pinpoint the primary sources of uncertainty, an analysis method based on uncertainty contribution factors was introduced, effectively identifying the most influential isotopes, reaction types, and energy regions. The sensitivity and uncertainty quantification analysis was performed on typical fast reactor systems, such as ABTR [39], CiADS [40], and MET-1000 [41], using covariance data from different nuclear data evaluation libraries.

The results indicate that β{eff} *is sensitive to both the average numbers of delayed neutrons (_d) and prompt neutrons (_p). Moreover, significant discrepancies were observed in the uncertainty estimation of* β{eff} due to differences in covariance data across evaluation libraries. The lack of covariance information for delayed neutrons of important nuclides in ENDF/B-VIII.1 and JEFF-3.3 leads to a substantial underestimation of the uncertainty in β{eff}. *Further analysis reveals that the uncertainty in* β{eff} primarily originates from the _d and _p of heavy nuclides, particularly in the high-energy region.

II. Nuclear Data Uncertainty Propagation Method

Methods for propagating nuclear data uncertainties are broadly categorized into two groups: random sampling and deterministic approaches. The random sampling technique involves first drawing a large number of samples from the probability distributions of all uncertain input parameters, and then substituting these sampled values into the model to obtain the corresponding output results. This method involves repeated execution of transport calculations using nuclear data sampled from covariance matrices. Statistical analysis of the transport calculation results assesses the propagated nuclear data uncertainty. While the principle is straightforward, the requirement for numerous repeated transport calculations necessitates significant computational resources and time. The deterministic approach involves first calculating the sensitivity coefficients of the output response and subsequently combining them with the input parameters' covariance matrix to determine the output uncertainty. Characterized by its computational efficiency, this method has become the mainstream technique for nuclear data uncertainty analysis.

The output parameters R in reactor physics are expressed as a function of nuclear data, and the uncertainty propagation of nuclear data in the deterministic approach follows the theoretical framework known as the "Sandwich Rule" :

$$SD(R) = \mathbf{S} \times \mathbf{Cov}_{\sigma} \times \mathbf{S}^{T}$$

where $SD(R)$ denotes the standard deviation of the output response R; σ is the nuclear data; $\mathbf{S}$ and $\mathbf{S}^T$ are its sensitivity coefficient vector and its transpose, respectively; and $\mathbf{Cov}_{\sigma}$ is the $N \times N$ covariance matrix of the nuclear data, with N representing the total number of nuclear data.

A. Production of Nuclear Data Covariance

Nuclear data serve as fundamental input parameters for reactor physics calculations, primarily obtained through experimental measurements and theoretical modeling. However, due to experimental uncertainties and approximations in theoretical models, inherent uncertainties exist in nuclear data [42–44]. These uncertainties are generally stored in the form of covariance matrices within evaluated nuclear data libraries. With the advancement of nuclear technology, particularly the deepening research into advanced nuclear energy systems and nuclear fuel cycles, the demand for nuclear data covariance information has become increasingly urgent. In 1979, ENDF/B-V released neutron cross section covariance data for ^{235}U and ^{238}U. Subsequently, ENDF/B-VI.8 systematically established a covariance evaluation framework, incorporating more fission nuclides and structural material nuclides, covering covariance data for neutron reaction cross-sections of 47 nuclides. The number of nuclides with covariance data in ENDF/B-VII.1 has increased to 190, greatly meeting the uncertainty analysis needs of various nuclide data. Simultaneously, other nuclear data evaluation libraries are also expanding their covariance coverage: JENDL-5.0 covers nearly 100 nuclides, while JEFF-3.3 exceeds 400, jointly promoting the development of nuclear data covariance research. The covariance data from nuclear data libraries are not formatted for direct use in uncertainty propagation. The process of group collapsing is therefore essential to convert these data into applicable multi-group covariance matrices.

In this paper, the multi-group cross-section covariance libraries, compatible with the neutronics calculation' s energy group structure, were generated using the nuclear data processing code NJOY2016, and the process is shown in Fig. 1 [Figure 1: see original paper].

B. Calculation of Sensitivity Coefficient

Sensitivity coefficient is an indicator used to quantify the sensitivity of a system' s output (R) to changes in input parameters (α). It describes the magnitude of the impact of small changes in α on R through mathematical methods. The specific expression is as follows:

$$S_{R,\alpha} = \frac{\partial R/R}{\partial \alpha/\alpha}$$

Although the differential operator and direct numerical perturbation methods are also used, the adjoint-weighted perturbation method is the predominant technique employed in sensitivity analysis, especially for calculating the sensitivity coefficients of k_{eff} with respect to nuclear data. Based on the relationship between $\beta\{eff\}\ and\ k$\{eff\}, this study extends the adjoint-weighted perturbation method to compute sensitivity coefficients of β_{eff} to nuclear data.

In nuclear reactors, neutron multiplication depends not only on _p but also on _d. The multiplication factor considering all neutrons (_p + _d) is referred to as the k_{eff}, while that considering only prompt neutrons is called the prompt neutron multiplication factor (k_p). The ratio of k_p to k_{eff} (k_p/k_{eff}) indicates the significance of delayed neutrons in maintaining reactor criticality. According to reactor dynamics theory, the relationship can be expressed as:

$$k_p = k_{\text{eff}}(1 - \beta_{\text{eff}})$$

The β_{eff} is calculated by:

$$\beta_{\text{eff}} = 1 - \frac{k_p}{k_{\text{eff}}}$$

Differentiating Eq. (4) with respect to the nuclear data yields the following expression for the sensitivity coefficient of β_{eff} with respect to the nuclear data:

$$S_{\beta_{\text{eff}},\sigma} = \frac{1}{1 - \beta_{\text{eff}}} \left(S_{k_{\text{eff}},\sigma} - S_{k_p,\sigma} \right)$$

where $S_{k_{\text{eff}},\sigma}$ is the relative sensitivity coefficient of k_{eff} to σ; $S_{k_p,\sigma}$ is referred to as the relative sensitivity coefficient of k_p to σ.

C. Identification of Uncertainty Sources

Experimental measurements constitute a cornerstone for improving the accuracy of nuclear data. However, comprehensive experimental optimization is neither practical nor economical due to high costs. Therefore, it is essential to first analyze the sources of nuclear data uncertainties and quantify the impact of different nuclear data on output parameters. Based on such quantitative analysis, optimization priorities can be established, focusing on the nuclear data that most significantly influence output parameters and guiding targeted experimental measurements. A new analytical approach, the uncertainty contribution factor method, is presented to quantify and rank the contributions of specific nuclear data to the total uncertainty. For this purpose, the uncertainty contribution factor is defined as follows:

$$\Delta_{nx} = \frac{\mathbf{S}_{\sigma_{n,x}} \times \mathbf{Cov}_{\sigma_{n,x}} \times \mathbf{S}^T_{\sigma_{n,x}}}{\sum_{n=1}^{N} \sum_{x=1}^{X} \mathbf{S}_{\sigma_{n,x}} \times \mathbf{Cov}_{\sigma_{n,x}} \times \mathbf{S}^T_{\sigma_{n,x}}}$$

where n denotes the nuclide index with N being the total number of nuclides; x represents the nuclear data type index with X being the total number of types; $\mathbf{S}_{\sigma_{n,x}}$ denotes the sensitivity coefficient vector of the output parameter with respect to the nuclear parameter x of nuclide n, $\mathbf{Cov}_{\sigma_{n,x}}$ is the variance of the nuclear parameter x of nuclide n; Δ_t represents the total uncertainty introduced by all nuclear parameters, while Δ_{nx} represents the uncertainty introduced by the nuclear parameter x of nuclide n.

To further decompose the uncertainty contributions of various parameters as a function of energy, an energy-dependent uncertainty contribution factor is introduced, which is defined as follows:

$$\Delta_{nxg} = \frac{\mathbf{S}_{\sigma_{n,x,g}} \times \mathbf{Cov}_{\sigma_{n,x,g}} \times \mathbf{S}^T_{\sigma_{n,x,g}}}{\sum_{n=1}^{N} \sum_{x=1}^{X} \sum_{g=1}^{G} \mathbf{S}_{\sigma_{n,x,g}} \times \mathbf{Cov}_{\sigma_{n,x,g}} \times \mathbf{S}^T_{\sigma_{n,x,g}}}$$

where g is the energy group index, G is the total number of groups. $\mathbf{S}_{\sigma_{n,x,g}}$ represents the sensitivity coefficient vector of R with respect to σ{n,x,g}*, and* $\mathbf{Cov}_{\sigma_{n,x,g}}$ *signifies the covariance associated with parameter* σ{n,x,g}.

The overall uncertainty quantification procedure for β_{eff} is illustrated in Fig. 2 [Figure 2: see original paper].

III. Computational Code Development

A computational code (MCSU) has been developed to quantify how nuclear data uncertainties propagate through β{eff} *calculations, and its computational workflow is shown in Fig. 2. The latest nuclear data evaluations from ENDF/B-VIII.1, JENDL-5.0, and JEFF-3.3 were processed using the NJOY2016 code to generate three dedicated 33-group covariance libraries. The 33-group sensitivity coefficients of* β{eff} with respect to nuclear data were determined by adjoint-weighted perturbation theory. The β_{eff} uncertainty introduced by nuclear data was subsequently quantified through convolution of these sensitivities with the 33-group nuclear data covariance matrix. In addition, an uncertainty contribution analysis method was proposed to systematically pinpoint the primary sources of uncertainty across specific isotopes, reaction types, and energy regions. The sensitivity and uncertainty quantification analyses were conducted on several typical fast reactor systems—ABTR, CiADS, and MET-1000—using covariance data from different evaluation libraries.

To evaluate the applicability of the computational code in fast reactor analysis, this study employs a pin-cell model based on the Sodium-cooled Fast Reactor (SFR) benchmark problem. The pin-cell geometry of the SFR is illustrated in Fig. 3 [Figure 3: see original paper]. In the hexagonal pin-cell configuration, a cylindrical fuel rod is surrounded by sodium coolant. The fuel composition consists of uranium-transuranics (U-TRU) with zirconium and natural molybdenum. Detailed specifications can be found in Reference [45]. Uncertainty on k_{inf} is calculated with covariance data from the ENDF/B-VII.1 evaluated nuclear data library, and the results are summarized in Table 1 . The first column lists results computed by the UNICORN code [46], developed at Xi' an Jiaotong University, while the second column presents results from the MCS code developed at UNIST. The comparison shows good agreement between the computational results.

IV. Numerical Results and Discussion

A. Advanced Burner Test Reactor (ABTR)

The primary mission of the ABTR is to demonstrate the transmutation of transuranics recovered from Light Water Reactor (LWR) spent fuel. The ABTR reference core design operates at a thermal power of 250 MWt, employing weapons-grade plutonium (WG-Pu) driver fuel with ternary metal alloy fuel, and uses liquid sodium as the coolant. The reactor core consists of 199 hexagonal assemblies arranged in concentric rings, as shown in Fig. 4 [Figure 4: see original paper]. It includes 60 fuel assemblies, 7 primary control rod assemblies, 3 secondary control assemblies, 9 material test assemblies, 78 reflector assemblies, and 48 shield assemblies. Detailed geometric dimensions and material compositions are provided in Reference [39].

Based on the ABTR model, a sensitivity and uncertainty analysis of β{*eff*} *was performed. Energy-integrated sensitivity coefficients of* β{eff} to nuclear data were calculated, and the most significant nuclear data sensitivity coefficients are presented in Fig. 5 [Figure 5: see original paper]. As observed in the figure, β{*eff*} *exhibits the highest sensitivity to the* _*p of* ^{239}*Pu, followed by the* _*d of* ^{238}*U, and then that of* ^{239}*Pu. According to the definition of* β{eff} (Eq. (4)), its value depends on both k_p and k_{eff}, which characterize the neutron multiplication performance of the system. Due to the strong correlation between β{*eff*} *and the average number of fission neutrons, the corresponding sensitivity coefficients are relatively large. Furthermore, it can be derived from Equation (4) that* β{eff} is negatively correlated with _p and positively correlated with _d.

To investigate the sensitivity of β{*eff*} *to nuclear data across specific energy ranges, energy-dependent sensitivity coefficients were computed using a 33-group energy structure. Figure 6 [Figure 6: see original paper] presents the 33-group sensitivity coefficients of key nuclear data for several major isotopes. It can be seen that the sensitivity of* β{eff} to the average numbers of _p and _d, and fission cross-section of ^{238}U is predominantly concentrated around an energy of 1 MeV. This energy dependence arises from the threshold fission behavior of ^{238}U, which requires incident neutron energies above approximately 1 MeV. In addition, the sensitivity of β_{eff} to the _p and _d of ^{239}Pu is mainly confined to the energy range from 0.01 MeV to 1.0 MeV.

Based on the 33-group sensitivity coefficients and the corresponding covariance matrices generated by NJOY2016 using evaluated nuclear data libraries—including ENDF/B-VIII.1, JENDL-5.0, and JEFF-3.3—the uncertainty in β{*eff*} *due to nuclear data was quantified. Table 2 presents the uncertainty in* β{eff} for the ABTR and the principal contributing sources. When using the ENDF/B-VIII.1 library, the uncertainty in β{*eff*} *is 1.67%. The largest contributor to the* *uncertainty in* β\{eff\} is the ^{52}Cr-σe *reaction cross-section, followed by the* ^{238}U-σ\{in\} reaction cross-section, and then the _d of ^{235}U. With the JENDL-5.0 library, the uncertainty of β*\{eff\} is 3.06%, with the dominant contributor being the _d of* ^{239}Pu*, followed by the _d of* ^{238}U*, and then the* ^{238}U-σ\{in\} reaction cross-section. For the JEFF-3.3 library, the β*\{eff\} uncertainty of 1.65% is primarily driven by the* ^{239}Pu *fission spectrum, while the* ^{238}U-σ\{in\} and ^{238}U-σf *reaction cross-sections are identified as secondary sources. It can be observed that there are significant differences in the calculated* β\{eff\} uncertainty when using different evaluated nuclear data libraries. This variation can be attributed to two factors: on the one hand, inherent differences exist in the nuclear data covariance among the libraries. Figure 7 [Figure 7: see original paper] presents the 33-energy-group covariance of the ^{238}U- _d based on ENDF/B-VIII.1 and JENDL-5.0. As seen, the covariance of the ^{238}U- _d in JENDL-5.0 is significantly larger than that in ENDF/B-VIII.1, which directly leads to its greater contribution to β*\{eff\} uncertainty in JENDL-5.0-based calculations. On the other hand, the ENDF/B-VIII.1 and JEFF-3.3 libraries lack complete covariance data for certain nuclides—for instance, neither includes covariance data for the _d of* ^{239}Pu*. Such omissions result in an underestimation of* β\{eff\} uncertainty when these libraries are used.

To pinpoint the key energy groups for nuclear data refinement aimed at reducing the uncertainty in β*\{eff\}, the energy-dependent contributions to this uncertainty were analyzed using the JENDL-5.0 library, as shown in Fig. 8 [Figure 8: see original paper]. The results indicate that the* β\{eff\} uncertainty arising from the ^{239}Pu- _d primarily originates in the energy range of 0.01-0.50 MeV, corresponding to energy groups 19 to 27. For the ^{238}U- _d, a significant contribution to β*\{eff\} uncertainty is observed between 0.5 MeV and 10.0 MeV, covering energy groups 27 to 32. Moreover, the uncertainty contribution of the* ^{238}U *inelastic scattering cross-section to* β\{eff\} occurs primarily in the 0.05–5.00 MeV energy range (groups 27-32), while that of the ^{239}Pu fission spectrum is predominantly from 0.05 MeV to 10 MeV (also groups 27–32).

B. Medium-size Metallic Core (MET-1000)

The MET-1000, designed by Argonne National Laboratory, is a 1000 MWth sodium-cooled fast reactor that utilizes metallic fuel and features a core comprising 379 assemblies. These include 180 fuel assemblies (in inner and outer zones), 114 radial reflector assemblies, 66 radial shield assemblies, and 19 control assemblies. Each assembly has a flat-to-flat distance of 16.2471 cm, and the total core height is 480.2 cm. The lattice configuration within each assembly and the radial layout of the 379 core assemblies are illustrated in Fig. 9 [Figure 9: see original paper]. The fuel is composed of a U–Pu–10Zr ternary metal alloy, the control rods primarily consist of B_4C, and the ducts and cladding are made of HT-9 alloy. Detailed geometric dimensions and material compositions of the model are provided in Reference [41].

Based on the MET-1000 model, a sensitivity and uncertainty analysis of β*\{eff\}*

was carried out. Energy-integrated sensitivity coefficients of β{eff} to nuclear data were computed, with the most significant ones illustrated in Fig. 10 [Figure 10: see original paper]. As shown in the figure, *β{eff} displays the highest sensitivity to the _p of ^{239}Pu, followed by the _d of ^{238}U and then that of ^{239}Pu. To further examine the energy-dependent sensitivity of* β{eff} to nuclear data, the sensitivity coefficients were computed using the 33-group energy structure. Figure 11 [Figure 11: see original paper] displays the resulting group-wise sensitivity coefficients for key nuclear data of several major nuclides. It can be observed that the sensitivity of β_{eff} to the _p, _d, and σ_f of ^{238}U is strongly concentrated around 1 MeV, which typically exhibits a fission threshold between 1 MeV and 2 MeV. Similarly, the sensitivity to both the _p and _d of ^{239}Pu is mainly focused within the energy range of 1–2 MeV.

Based on the 33-group sensitivity coefficients and the corresponding covariance matrices, the uncertainty in *β{eff} was quantified. Table 3 summarizes the resulting* β{eff} uncertainty for the MET-1000 system and its main contributing sources. When the ENDF/B-VIII.1 library is employed, the resulting β_{eff} uncertainty reaches 2.05%. This uncertainty is dominated by the ^{54}Fe-σ_e reaction cross-section, with significant secondary contributions from the ^{23}Na-*σe and ^{56}Fe-σ*e reactions. With the JENDL-5.0 library, the *β{eff} uncertainty increases to 3.24%, dominated by the ^{239}Pu- _d, followed by the fission spectrum of ^{239}Pu and then the ^{238}U- _d. In the case of JEFF-3.3, the* β{eff} uncertainty is 1.84%, mainly originating from the fission spectrum of ^{239}Pu, followed by the ^{56}Fe-*σe reaction cross-section and the ^{23}Na-σ*e reaction cross-section. As can be seen from Table 3, there are significant differences in the calculated *β{eff} uncertainties when using covariance data from different nuclear data evaluation libraries. Figure 12 [Figure 12: see original paper] shows the covariance of the ^{239}Pu fission spectrum from different evaluated libraries. Among them, the covariance in the JEFF-3.3 library is noticeably larger than that in ENDF/B-VIII.1 and JENDL-5.0, thereby contributing the most to the overall uncertainty. Furthermore, it can be observed from Fig. 13 [Figure 13: see original paper] that the contribution of the ^{239}Pu fission spectrum to* β{eff} is primarily concentrated in the higher energy region. Consequently, the uncertainty contribution of the ^{239}Pu fission spectrum in JENDL-5.0 is greater than that in ENDF/B-VIII.1, which is closely associated with the larger covariance of its spectral data in the high-energy region.

The contribution of nuclear data uncertainties to *β{eff} was analyzed based on JENDL-5.0, as shown in Fig. 13. The* β{eff} uncertainty induced by the ^{239}Pu- _d originates mainly in the energy range of 0.50 keV to 0.20 MeV, corresponding to energy groups 13–25. For the fission spectrum of ^{239}Pu, a significant contribution to *β{eff} uncertainty lies in the 0.05–5.0 MeV range, covering groups 22–32. Meanwhile, the uncertainty contribution of the ^{238}U- _d to* β{eff} stems mainly from 0.80 MeV to 5.00 MeV (groups 28–32). In contrast, the uncertainty contribution of the ^{56}Fe-σ_e derives predominantly from 100 eV to 2 MeV (energy groups 10–30).

C. China Initiative Accelerator Driven System (CiADS)

The CiADS is a megawatt-class ADS principle verification facility led by the Chinese Academy of Sciences. Its primary objective is to demonstrate the system integration of a superconducting linear accelerator, a subcritical reactor, and a spallation target, thereby validating the fundamental principles of minor actinide (MA) transmutation. The CiADS subcritical reactor has a thermal power of 10 MW, with an average power density of 38.49 W/cm^3 and an average linear power density of 39.97 W/cm. The CiADS uses a lead-bismuth target, and the subcritical core is composed of fuel assemblies, reflection assemblies, and shielding assemblies, as shown in Fig. 14 [Figure 14: see original paper]. The fuel assemblies are configured with 61 rods of UO_2 at 19.75% enrichment. The reflector and shielding assemblies, in contrast, employ 7 stainless steel rods and 7 boron carbide (B_4C) rods, respectively.

Figure 15 [Figure 15: see original paper] illustrates the dominant energy-integrated sensitivity coefficients of β{*eff*} *with respect to nuclear data, obtained from a comprehensive sensitivity and uncertainty analysis of the CiADS model. It can be observed that* β{eff} shows the highest sensitivity to the __d of ^{235}U, followed by the ^{235}U-σf*, and then the* ^{237}*Np-__p. To further investigate the energy-dependent sensitivity of* β{eff} to nuclear data, the sensitivity coefficients were computed using the 33-group energy structure. Figure 16 [Figure 16: see original paper] presents the resulting group-wise sensitivity coefficients for key nuclear data of several major nuclides. The sensitivity of β_{eff} to both the ^{235}U- __p and ^{235}U-σ_f is predominantly concentrated in the energy range from 1 keV to 1 MeV. Similarly, the sensitivity to the __p and σ_f of ^{237}Np is mainly localized around 1 MeV.

Table 4 presents the uncertainty in β{*eff*} *for the CiADS and the principal contributing sources. When using the ENDF/B-VIII.1 library, the uncertainty in* β{eff} is 4.06%. The dominant contributor to the uncertainty in β{*eff*} *is the* ^{235}*U- __d, with subsidiary roles played by the* ^{16}*O-*σe and ^{235}U-σf *reaction cross-sections. With the JENDL-5.0 library, the uncertainty of* β{eff} is 3.55%, with the dominant contributor being the __d of ^{235}U, followed by the fission spectra of ^{241}Am, and then the ^{56}Fe-σe *reaction cross-section. An uncertainty of 2.37% in* β{eff} was observed with the JEFF-3.3 library, which is largely attributed to the ^{235}U-σf *reaction cross-section. The* σf and $\sigma\gamma$ *reactions of* ^{237}*Np, along with the* ^{56}*Fe-*σe reaction, are ranked as the next most significant contributors. It can be observed that there are significant differences in the calculated β{*eff*} *uncertainty when using different evaluated nuclear data libraries. The uncertainty in* β{eff} mainly originates from ^{235}U, which is closely related to the use of 19.75% enriched ^{235}U as the fuel in CiADS. Due to the lack of delayed neutron covariance data for ^{235}U in the JEFF-3.3 library, the β{*eff*} *uncertainty calculated using JEFF-3.3 covariance data is significantly lower than that obtained with JENDL-5.0 and ENDF/B-VIII.1. Notably, in the two previous fast reactor cases, the* β{eff} uncertainty calculated based on JENDL-5.0 was greater than that based on ENDF/B-VIII.1, whereas the CiADS results show the opposite trend. This discrepancy is primarily attributed to the significantly larger covariance of the _d for ^{235}U in ENDF/B-VIII.1 compared to that in JENDL-5.0, as shown in Fig. 17 [Figure 17: see original paper].

To further support nuclear data refinement and reduce the resulting uncertainty in β{eff}, *the contribution of nuclear data uncertainties to* β{eff} was analyzed using JENDL-5.0, as shown in Fig. 18 [Figure 18: see original paper]. The results indicate that the β{eff} *uncertainty arising from the _d of* ^{235}U *primarily originates in the 0.02–0.50 MeV range, corresponding to energy groups 21–27. For the fission spectrum of* ^{241}Am*, a significant contribution to* β{eff} uncertainty lies in the 0.10 MeV and 5.0 MeV range, covering energy groups 25 to 32. Moreover, the uncertainty in β_{eff} arising from the ^{56}Fe-σe *reaction is predominantly located in the 0.10 keV–0.10 MeV range (groups 11–27), while that of the* ^{56}Fe-σ{in} reaction cross-section is concentrated mainly in the 0.50–2.0 MeV range (groups 28–30).

V. Summary and Conclusion

Multi-group covariance libraries were first generated from the latest nuclear data evaluations (ENDF/B-VIII.1, JENDL-5.0, JEFF-3.3) using NJOY2016. Subsequently, a methodology for quantifying nuclear data uncertainty propagation in β{eff} *calculations was established based on these libraries. Multi-group sensitivity coefficients of nuclear data to* β{eff} were computed by applying adjoint-weighted first-order perturbation theory. Furthermore, an uncertainty contribution factor method was developed to pinpoint the dominant sources of uncertainty across specific isotopes, reaction types, and energy regions. Sensitivity and uncertainty quantification analyses were performed on typical fast reactor systems, including ABTR, MET-1000, and CiADS. The principal conclusions derived from the computational analysis are as follows:

(1) β{eff} *exhibits sensitivity to several nuclear parameters, including the average number of delayed neutrons, the average number of prompt neutrons, and the fission cross-section. Both the ABTR and MET-1000 models exhibit consistent sensitivity profiles for* β{eff}, showing dominant sensitivity to the average number of prompt neutrons of ^{239}Pu, with the average number of delayed neutrons of ^{238}U being the secondary contributor. For the CiADS model, β{eff} *exhibits dominant sensitivity to the average number of delayed neutrons of* ^{235}U*, with a secondary influence from its fission cross-section. This difference arises from the different fuel types used: the ABTR and MET-1000 models use U-Pu-Zr alloy fuel, while the CiADS model uses highly enriched* ^{235}U *fuel. Meanwhile, the sensitivity coefficients of* β{eff} with respect to the number of delayed neutrons, prompt neutrons, and fission cross-section are all characterized by prominent peaks near the 1 MeV energy range.

(2) Significant discrepancies are observed in the uncertainty estimates of β{eff} *based on the covariance matrices from the ENDF/B-VIII.1,*

JENDL-5.0, and JEFF-3.3 nuclear data libraries. The resulting uncertainties in β{eff} for each system are as follows: ABTR model: 1.67%, 3.06%, and 1.65%, respectively; MET-1000 model: 2.05%, 3.24%, and 1.84%, respectively; CiADS model: 4.06%, 3.55%, and 2.37%, respectively. It is evident that the uncertainties in *β{eff} obtained from the ENDF/B-VIII.1 and JEFF-3.3 libraries are significantly lower than those computed using the JENDL-5.0 library. This discrepancy is primarily due to the lack of covariance data for the delayed neutron yields of certain nuclides in both the ENDF/B-VIII.1 and JEFF-3.3 libraries. However, in the CiADS model, the uncertainty in* β{eff} based on the ENDF/B-VIII.1 library is actually higher than that derived from JENDL-5.0. This can be attributed to the fact that in the CiADS model, the uncertainty in *β{eff} mainly originates from ^{235}U. Furthermore, the covariance value for the delayed neutron yield of ^{235}U in the ENDF/B-VIII.1 library is considerably larger than the corresponding value in the JENDL-5.0 library, resulting in a greater contribution to the overall uncertainty in* β{eff} during uncertainty propagation.

(3) The uncertainty contribution factor method was proposed to investigate the sources of uncertainty in *β{eff}. Owing to the comprehensive covariance data available in the JENDL-5.0 library, the analysis is primarily conducted based on results computed from this library. The analysis reveals significant differences in the key nuclides, reaction types, and energy ranges contributing to* β{eff} uncertainty across different models: In the ABTR model, the uncertainty in β_{eff} is mainly attributed to the ^{239}Pu- _d and ^{238}U- _d. Specifically, the ^{239}Pu- _d contribution is concentrated in the 0.01–0.50 MeV range, while that of ^{238}U- _d is primarily distributed between 0.5-10.0 MeV. In the MET-1000 model, the uncertainty is predominantly influenced by the ^{239}Pu- _d and fission spectrum of ^{239}Pu. The ^{239}Pu- _d contribution lies mainly in the 0.50–0.20 MeV range, whereas that of the ^{239}Pu fission spectrum is concentrated in the 0.05–5.0 MeV region. In the CiADS model, the uncertainty stems primarily from the ^{235}U- _d and the fission spectrum of ^{241}Am. The ^{235}U- _d contribution is mainly in the 0.02–0.50 MeV range, and that of the ^{241}Am fission spectrum is concentrated between 0.10–5.0 MeV.

VI. Author Contributions

All authors contributed to the study conception and design. Material preparation, numerical modeling, computational analysis, and the initial draft of the manuscript were carried out by CY. Data collection and modeling were performed by LL and C-FH. Literature review and data organization were conducted by L-LW. Z-PC provided critical revisions to the manuscript. YT contributed to the methodology and acquired funding. All authors reviewed and commented on previous versions of the manuscript, and read and approved the final manuscript.

VII. Bibliography

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Note: Figure translations are in progress. See original paper for figures.