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Original Article

Sensitivity and Uncertainty Analysis on Reactivity for HEU and LEU Fuel Assemblies of Dalat Nuclear Research Reactor using Monte Carlo Code and ENDF/B-VII.0 and ENDF/B-VII.1 Nuclear Libraries

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Abstract: This paper presents the analysis of sensitivity and uncertainty for the infinite multiplication factor (k_{inf}) for the VVR-M2 typed HEU and LEU fuel assemblies of the Dalat nuclear research reactor (DNRR) using the MCNP6.1-Whisper1.1 code. Sensitivity calculations were performed for the ENDF/B-VII.0 and ENDF/B-VII.1 nuclear data libraries. In Whisper, the k_{eff} uncertainty due to nuclear data was evaluated by the uncertainty propagation law using the sensitivities obtained by MCNP and the available covariance matrix. The most significant sensitivity coefficients in positive contribution are the coefficient total ν and fission reaction of U-235, elastic scattering reaction of H-1 and inelastic scattering of thermal neutrons and in negative contribution are capture reactions of U-235, H-1, Al-27, U-238 and U-234 isotopes. The large discrepancy between sensitivities of elastic scattering reaction cross section of H-1 and inelastic scattering of thermal neutrons between two libraries are found because of the change in neutron spectra of HEU and LEU fuel assemblies using the two library versions. The uncertainty of the k_{inf} from the

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ENDF/B-VII.0 nuclear library error for all isotopes was found to be significant (about 0.45% of reactivity effect) with the largest contribution from isotope U-235 contributions (about 0.3% to 0.40%).

Keywords: Sensitivity, Uncertainty, DNRR, MCNP6.1- Whisper-1.1, ENDF/B-VII.0, ENDF/B-VII.1.

1. Introduction

DNRR is the only nuclear research reactor in Vietnam at present. The reactor was built in the 1960s with an initial power of 250 kW. In the 1980s, the power of the reactor increased to 500 kW, using Russian VVR-M2 fuel type. Since the reactor was upgraded in 2007, the core of DNRR loaded by high-enriched uranium (HEU) fuel with U-235 enrichment up to 36% [1]. According to the reduced enrichment for research and test reactors (RERTR) Program, Dalat nuclear reactor converted fuel from HEU to low-enriched uranium (LEU) in 2007 and reached criticality for the first time with LEU fuel in 2011. Since then, the DNRR has operated stably with LEU fuel.

To evaluate the operation of the reactor core, researchers use simulation models. The physical characteristics of the DNRR core have been calculated using computational programs such as MCNP, WIMSD/CITATION, SRAC codes [2] and the reliability of the result depends directly on the accuracy of the cross section libraries. Thus, it is essential to do sensitivity and uncertainty analysis to evaluate the impact of nuclear data error on the calculation result. In this preliminary study, sensitivity and uncertainty analyses were performed for HEU and LEU fuel assemblies of the DNRR. The sensitivity calculation was performed using the MCNP6.1 code [3] for ENDF/B-VII.0 [4] and ENDF/B-VII.1 [5] nuclear data libraries and the uncertainty analysis was done using Whisper1.1 module [6] of MCNP6.1 code with the available SCALE format 44 energy group covariance cross section for ENDF/B-VII.0 [5] nuclear data library. The DNRR and its fuel assembly configuration are described in detail in Section 2. The calculation tools used for the analysis are provided in Section 3. The sensitivity and uncertainty analysis results are presented and discussed in Section 4. The conclusion and remarks are delivered in the final part of the paper.

2. Description for the DNRR Fuel Assembly Configurations

The DNRR is TRIGA Mark II reactor type with power upgraded to a 500 kW. The reactor structure consists of an aluminum water tank with 6 m in height and 2 m in diameter, placed between concrete blocks with a thickness of 2.5 m in the bottom half and 0.9 m in the top half. The core of the DNRR is cylindrical, 60 cm high, 44.2 cm in diameter, attached to a 2 m high suction well. The reactor core is surrounded by graphite and beryllium reflectors, which limit the leakage of neutrons from the core. To control the operation of the core, the DNRR has 2 safety rods, 4 shim rods and 1 automatic regulating rod. Safety rods and shim rods are made of B_4C, automatic regulating rod is made of stainless steel [1-2].

The reactor core contains 121 hexagonal lattice cells including fuel bundles, control rods, irradiation channels, and beryllium blocks. Prior to 2011, DNRR used VVR-M2 type high-enriched uranium fuel bundle (HEU-36% wt). Figure 1 shows a cross section of the VVR-M2 type HEU fuel bundle, which is made of an alloy of aluminum uranium cladded in aluminum. Each fuel assembly consists of 2 layers including a cylindrical inner layer and a hexagonal outer layer. Each fuel bundle contains approximately 40.2 g of U-235 mass [7]. According to the project named Russian Research Reactor Fuel Return (RRRFR) Program signed by the United States, the Russia Federation and the IAEA since 2004, the DNRR has also been converted from HEU to LEU since September 2007. The DNRR reached its criticality with LEU fuel for the first time in November 2011 with 72 LEU fuel bundles and was increased to 92 fuel bundles in December 2011. The LEU fuel bundle is enriched 19.75 wt.%, with an average U-235 mass of about 49.7 g. In this study, we utilize the same configuration of the HEU fuel assembly for LEU fuel assembly, only thickness of the fuel meat is increased to 0.94 mm and the clad thickness is reduced to 0.78 mm. The detailed parameters of HEU and LEU fuel rods are described in Table 1.

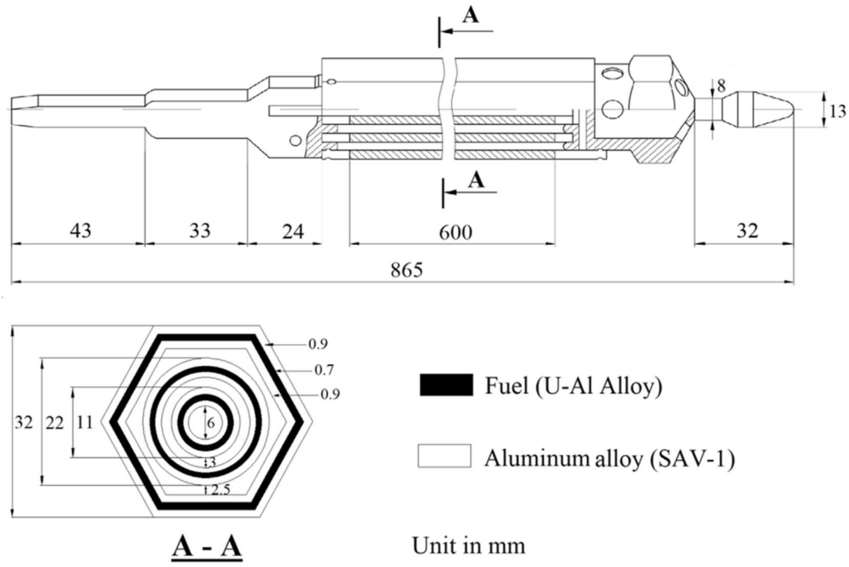

Figure 1. The VVR-M2 type HEU fuel bundle of DNRR [1]

Table 1. Parameters of HEU and LEU fuels [1]

Parameters	HEU	LEU
Number of elements in the fuel assembly	3	3
Hexagonal tube (outer fuel)	1	1
Cylindrical tube (inside fuel)	2	2
Fissionable material composition	U- Al	UO_2 + Al
Fuel element thickness (mm)	2,5	2,5
Fissionable material thickness in the fuel element (mm)	0,7	0,94
Cladding thickness (mm)	0,9	0,78
Gap between fuel elements (mm)	2,5÷3	2,5÷3
Total length (mm)	865	865
Nominal active length (mm)	600	600
Enrichment (%)	36	19,75
Average U-235 mass (g)	about 40,2	about 49,7
U-234	0,14	0,11
U-235	12,60	10,28
U-238	22,26	41,66
Al-27	65,00	40,93
O-16	-	7,02

3. Calculation Tools

3.1. MCNP6.1 Code

MCNP code (Monte Carlo N-Particle) is a program used to simulate the transport of radiation particles in three dimensions. This program can simulate the transport of 37 different types of particles for the purposes of criticality, shielding, dosimetry, detector response and many other applications. The Monte Carlo particle transport method has been practiced at the Los Alamos National Laboratory (LANL) since the 1940s. Since then, many versions of the MCNP program have been released. MCNP6.1 version was released in June 2013 with the ENDF/B-VII.1 nuclear data library. MCNP6 is a merger from 2 versions MCNP5 and MCNPX. Therefore, it contains features of both MCNP5 and MCNPX. In addition, MCNP6 has 16 new features added and many improved features [6]. In this study, sensitivity coefficient profile of k_{inf} with continuous energy physics for cross-section, fission multiplicities, spectrum and scattering distribution is used through the KSEN card in MCNP6.1 with 44 energy groups. Calculations were performed on both evaluated nuclear data libraries ENDF/B-VII.0 (2006) and ENDF/B-VII.1 (2011).

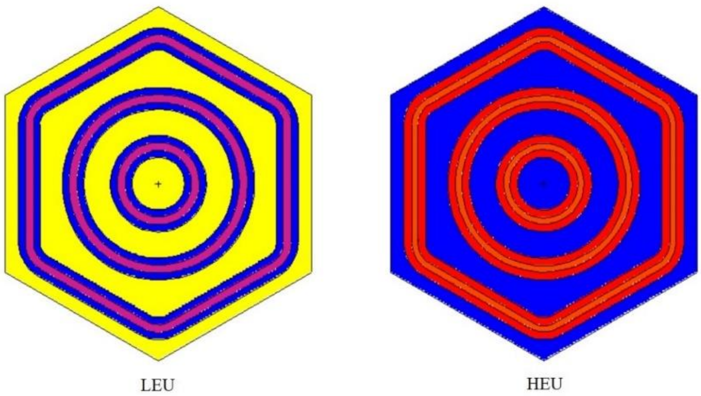

Figure 2. MCNP models of HEU fuel and LEU fuel assemblies

3.2. Whisper 1.1 Module

Whisper is a computational software for nuclear critical safety (NCS) analysis to determine the upper subcritical limit of the multiplying system based on the Monte Carlo calculation method of radiation transport. Whisper version 1.1 was released with the MCNP6.1 program [3, 8]. However, Whisper 1.1 can be used together with all versions of MCNP6, which have the same extensive verification validation testing for NCS. Uncertainty was calculated using Whisper software using 44-group covariance library of SCALE6.1 and sensitivity profiles were calculated from MCNP6 program using the “Sandwich Rule” [9]. In this study, we only use the Whisper program to calculate the uncertainty from the nuclear data library ENDF/B-VII.0.

4. Results and Discussion

The HEU and LEU fuel assemblies were modelled using MCNP6.1 code and their k_{inf} were obtained and listed in Table 2 and their neutron spectra are graphically presented in Figure 3 and Figure 4.

Table 2. K_{inf} of HEU and LEU fuel assemblies using ENDF/B-VII.0 and ENDF/B-VII.1

Fuel	HEU		LEU	
Nuclear data library	ENDF/B-VII.0	ENDF/B-VII.1	ENDF/B-VII.0	ENDF/B-VII.1
k_{inf}	1.63953 ±0.00003	1.63943 ±0.00003	1.63391±0.00003	1.63383±0.00003

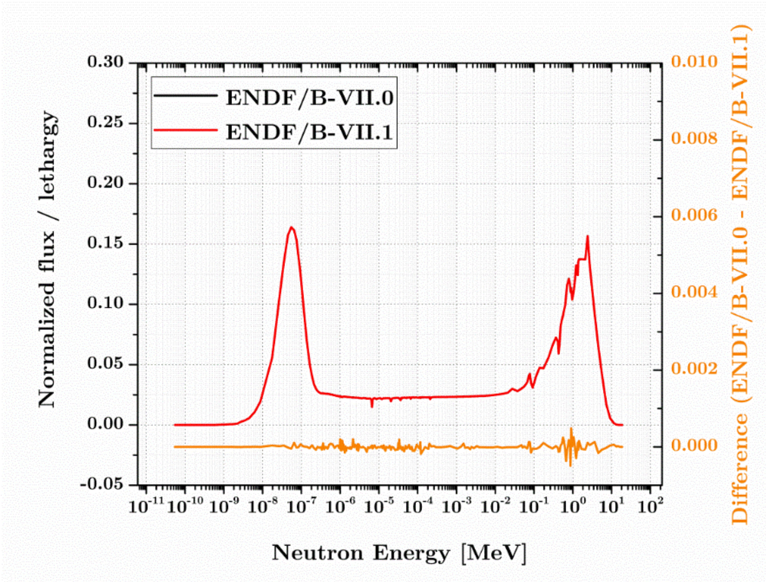

Figure 3. Neutron spectrum of HEU fuel assembly

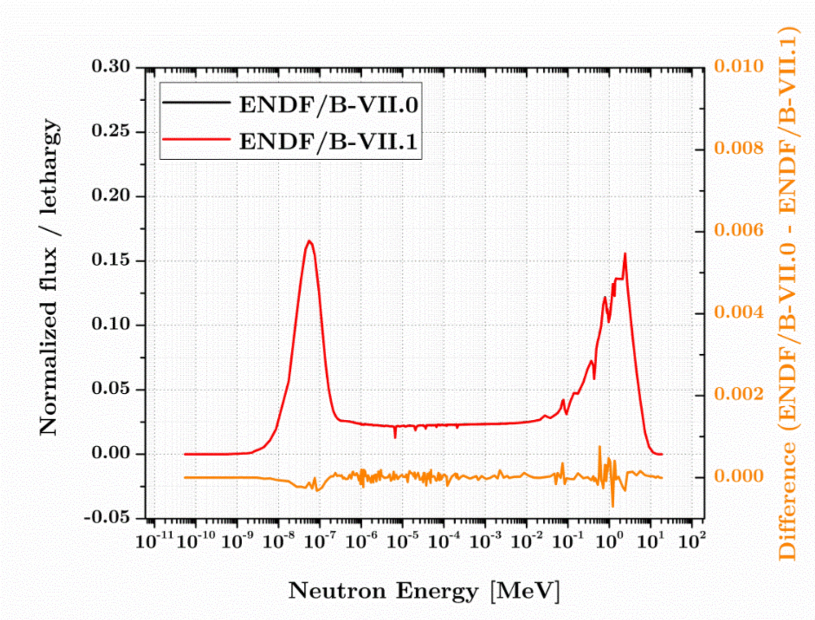

Figure 4. Neutron spectrum of LEU fuel assembly.

4.1. Sensitivity and Uncertainty of HEU Fuel

The calculated sensitivity coefficients of k_{inf} for each nuclide and reaction from ENDF/B-VII.0 and ENDF/B-VII.1 nuclear data libraries are shown in Table 3. Only the sensitivity coefficients with absolute value >0.1% were listed and discussed. From Table 3, it can be seen that top 5 isotopes with the most significant positive sensitivity coefficient total ν and fission reaction of U-235, elastic scattering reaction of H-1, inelastic scattering of thermal neutron cross section of light water (Lwtr) and elastic scattering of U-238. Their energy-dependent sensitivity profiles are plotted in the same scale for convenient comparison as illustrated in Figure 5. It shows that the sensitivity coefficient of U-235 total ν and fission reaction have a sharp peak at thermal energy range (0.1 eV) while the sensitivity coefficient of elastic scattering reaction of H-1 is more significant at epi-thermal range. It is obvious that the sensitivity of thermal neutron scattering is significant at thermal range. It can be explained due to the direct dependence of the sensitivity on the cross section value. As we can see from Figure 7, the cross section of U-235 fission reaction has large magnitude at thermal energy which causes its sensitivity coefficient to be significant at this range, while the elastic scattering reaction cross section of H-1 is larger at the epi-thermal range than the fast neutron region. A good agreement in sensitivity results of ENDF/B-VII.0 and ENDF/B-VII.1 are observed for sensitivity coefficient total ν and fission reaction of U-235. An inconsistency in sensitivity of elastic scattering reactions of H-1 and inelastic scattering of thermal neutrons is found (Figure 5 and Table 3). It mostly concentrates in the same regions where the disparity of neutron spectrum is found (Figure 3). Thus, it can be concluded that the discrepancy in neutron spectrum using ENDF/B-VII.0 and ENDF/B-VII.1 causes the difference in sensitivity coefficient between two libraries.

Table 3. Major sensitivities of k_{inf} (positive direction) for HEU fuel assembly

Isotopes	Reaction	ENDF/B-VII.0	ENDF/B-VII.1
U-235	total ν	9.98E-01	9.98E-01
U-235	fission	2.81E-01	2.80E-01
H-1	elastic	2.31E-02	2.15E-02
Lwtr	inelastic	1.88E-02	2.01E-02
U-238	elastic	2.80E-03	2.80E-03
U-238	total ν	2.05-03	2.05E-03

Table 4. Major sensitivities of k_{inf} (negative direction) for HEU fuel assembly

Isotopes	Reaction	ENDF/B-VII.0	ENDF/B-VII.1
U-235	n-gamma	-1.40E-01	-1.40E-01
H-1	n-gamma	-1.05E-01	-1.05E-01
Al-27	n-gamma	-4.99E-02	-4.99E-02
U-238	n-gamma	-2.94E-02	-2.94E-02
U-234	n-gamma	-2.55E-03	-2.55E-03

For negative sensitivity coefficient, the top five significant sensitivities are the capture reactions of U-235, H-1, Al-27, U-238 and U-234 isotopes (see Table 4). As shown in the energy dependent sensitivity profiles (Figure 6), sensitivity coefficients of neutron capture reactions of H-1, U-235, Al-27 peak at about 0.1 eV because their cross section has large magnitude at thermal region (Figure 8). The sensitivities of capture reaction U-234 and U-238 peak at energies around several eV where the resonance region of their cross section is located. An excellent agreement of the sensitivity between two cross section libraries is found.

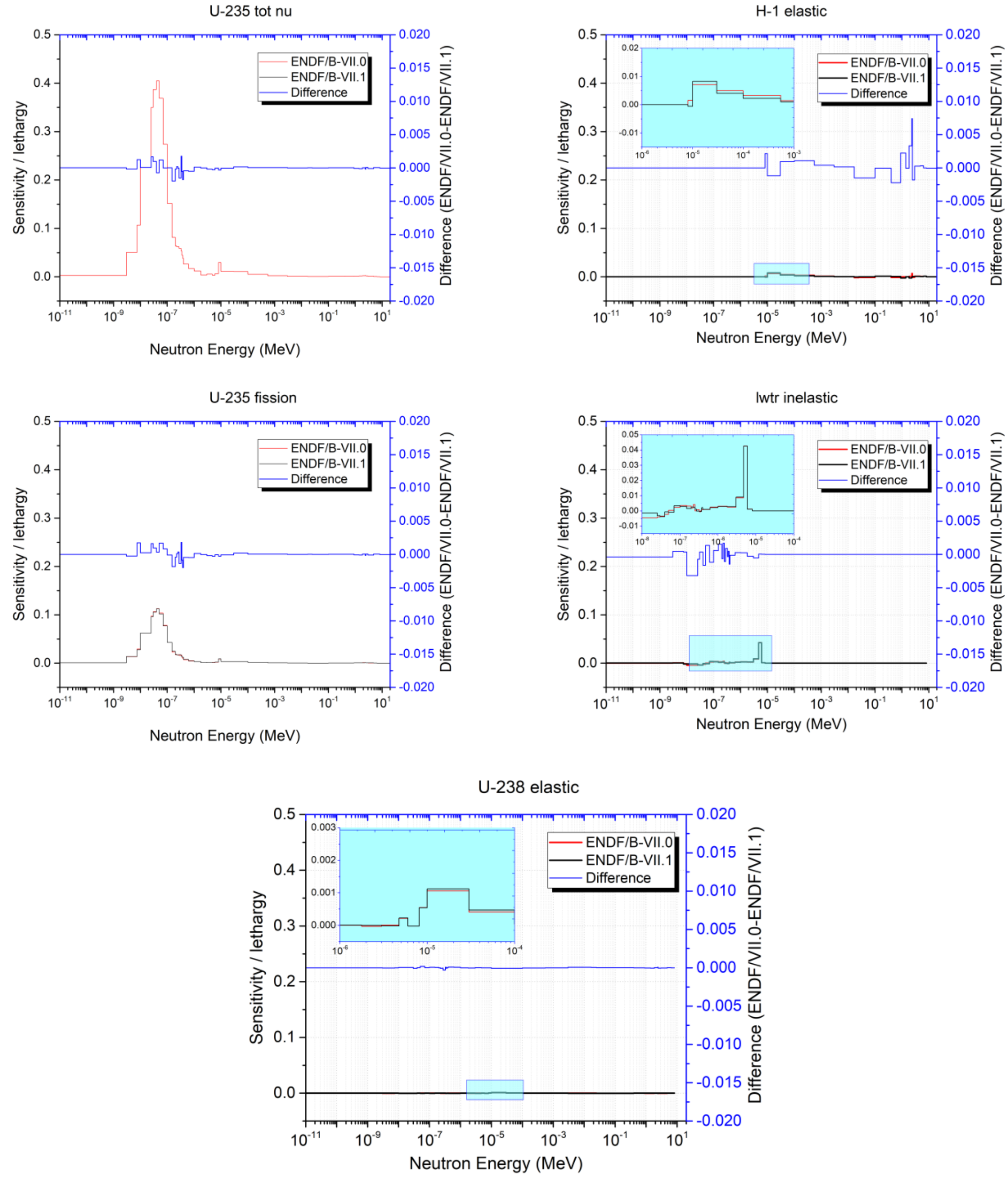

Figure 5. Energy dependent sensitivity coefficients of k_{inf} for HEU fuel assembly for ENDF/B-VII.0 and ENDF/B-VII.1 (positive contributions)

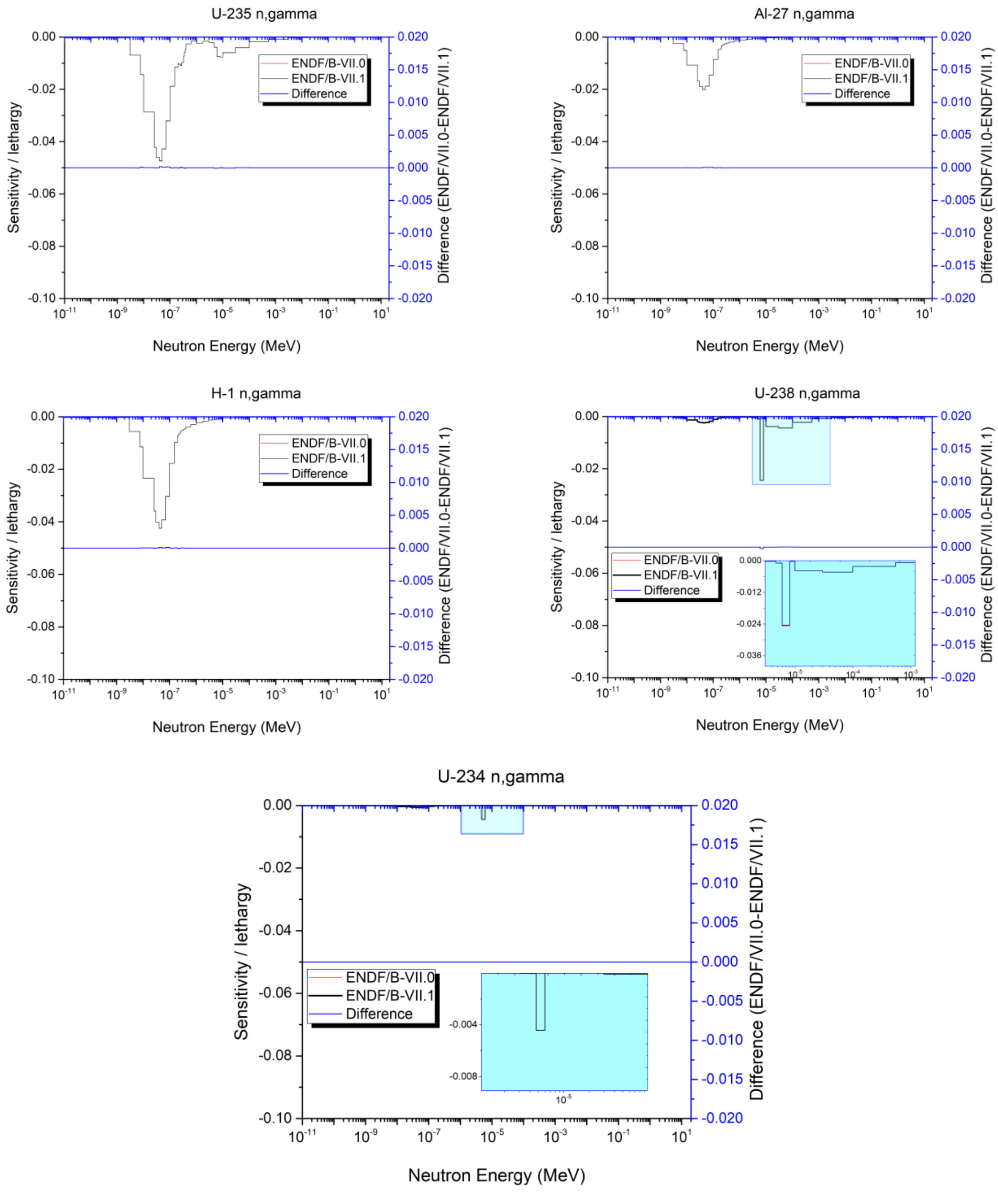

Figure 6. Energy dependent sensitivity coefficients of k_{inf} for HEU fuel assembly for ENDF/B-VII.0 and ENDF/B-VII.1 (negative contributions)

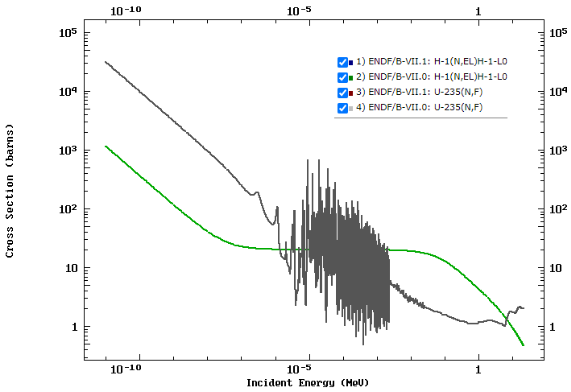

Figure 7. H-1 elastic scattering and U-235 fission cross section of ENDF/B-VII.0 and ENDF/B-VII.1libraries

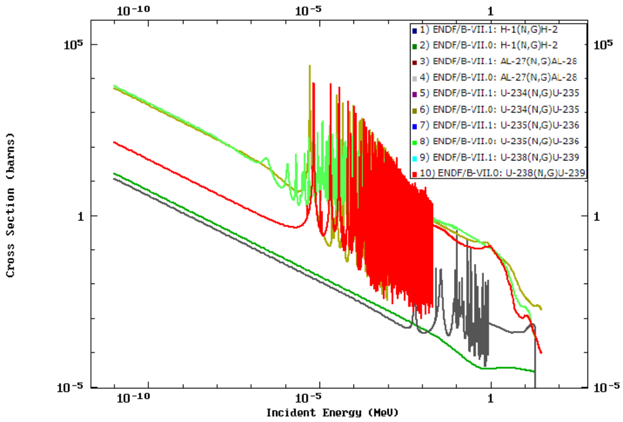

Figure 8. H-1. Al-27, U-234, U-235 and U-238 capture cross section
of ENDF/B-VII.0 and ENDF/B-VII.1libraries

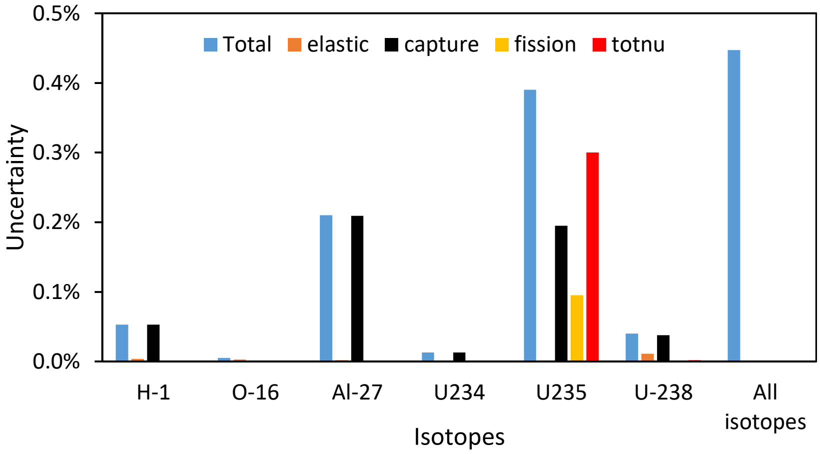

Figure 9. Uncertainty k_{inf} caused by ENDF/B-VII.0 library for HEU fuel assembly

The uncertainty of the isotopes in the HEU fuel assembly obtained by the Whisper1.1 module using available 44-group covariance matrix library of ENDF/B-VII.0 is depicted in Figure 9. The uncertainty for all the isotopes is 0.447% of the reactivity effect, with the major contribution from U-235, Al-27, H-1, U-238, U-234 and O-16 isotopes. The uncertainty caused by all reaction cross-sections of the U-235 isotope is 0.390%, the total ν reaction is the greatest contributor (0.300%), and the following are the neutron capture reaction (0.195%) and fission (0.095%). The uncertainty of the Al-27 isotope is mainly due to the neutron capture reaction (0.209%) compared to the total uncertainty from Al-27 (0.210%). The uncertainty from the reaction cross-section of H-1 is caused by elastic scattering (0.004%) and neutron capture (0.053%) reactions. The contribution of O-16, U-234 and U-238 isotopes is very small, 0.005%, 0.014% and 0.040% respectively.

4.2. Sensitivity and Uncertainty of LEU Fuel

Similar to the sensitivity of the HEU fuel assembly, the total ν reaction and fission of U-235 have the greatest positive sensitivity of k_{inf} of LEU fuel assembly with their peaks located at thermal energy region. Its energy-dependent sensitivity profile shows a small difference at thermal energy range (Figure 10) but the integrated sensitivity coefficient shows a good agreement (Table 5). The energy-dependent sensitivity strongly relies on the spectrum of the core. The changes in neutron spectrum lead to the change in sensitivity coefficient. The larger change in neutron spectrum of LEU (Figure 4) results in a larger discrepancy between sensitivities of elastic scattering reaction cross sections of H-1 and thermal neutron inelastic scattering between two libraries. On the other hand, for the LEU case, a very good consistency in negative sensitivity profiles and its integral sensitivity is observed in Figure 11 and Table 6.

Table 5. Major sensitivities of k_{inf} (positive direction) for LEU fuel assembly

Isotopes	Reaction	ENDF/B-VII.0	ENDF/B-VII.1
U-235	total ν	9.94E-01	9.94E-01
U-235	fission	2.63E-01	2.63E-01
H-1	elastic	3.58E-2	3.54E-02
Lwtr	inelastic	2.89E-02	2.66-02
U-238	total ν	5.71E-03	5.75E-03
U-238	elastic	4.55E-03	4.67E-03
U-238	fission	2.35E-03	2.38E-03

Table 6. Major sensitivities of k_{inf} (negative direction) for LEU fuel assembly

Isotopes	Reaction	ENDF/B-VII.0	ENDF/B-VII.1
U-235	n-gamma	-1.45E-01	-1.45E-01
H-1	n-gamma	-8.76E-02	-8.76E-02
U-238	n-gamma	-5.17E-02	-5.18E-02
Al-27	n-gamma	-3.69E-02	-3.69E-02
U-234	n-gamma	-2.83E-03	-2.84E-03

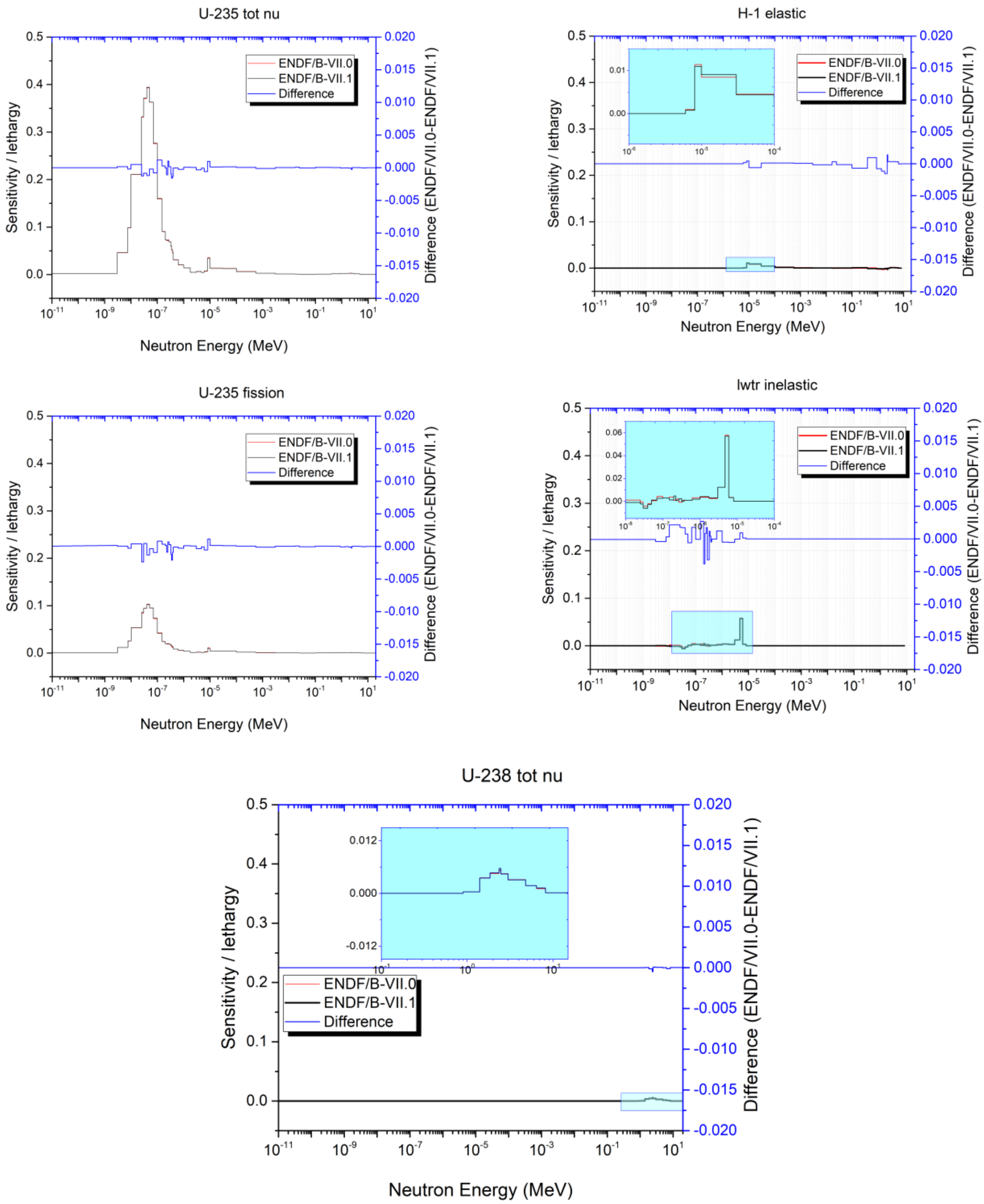

Figure 10. Energy dependent sensitivities of k_{inf} of LEU fuel assembly for ENDF/B-VII.1 (positive contributions).

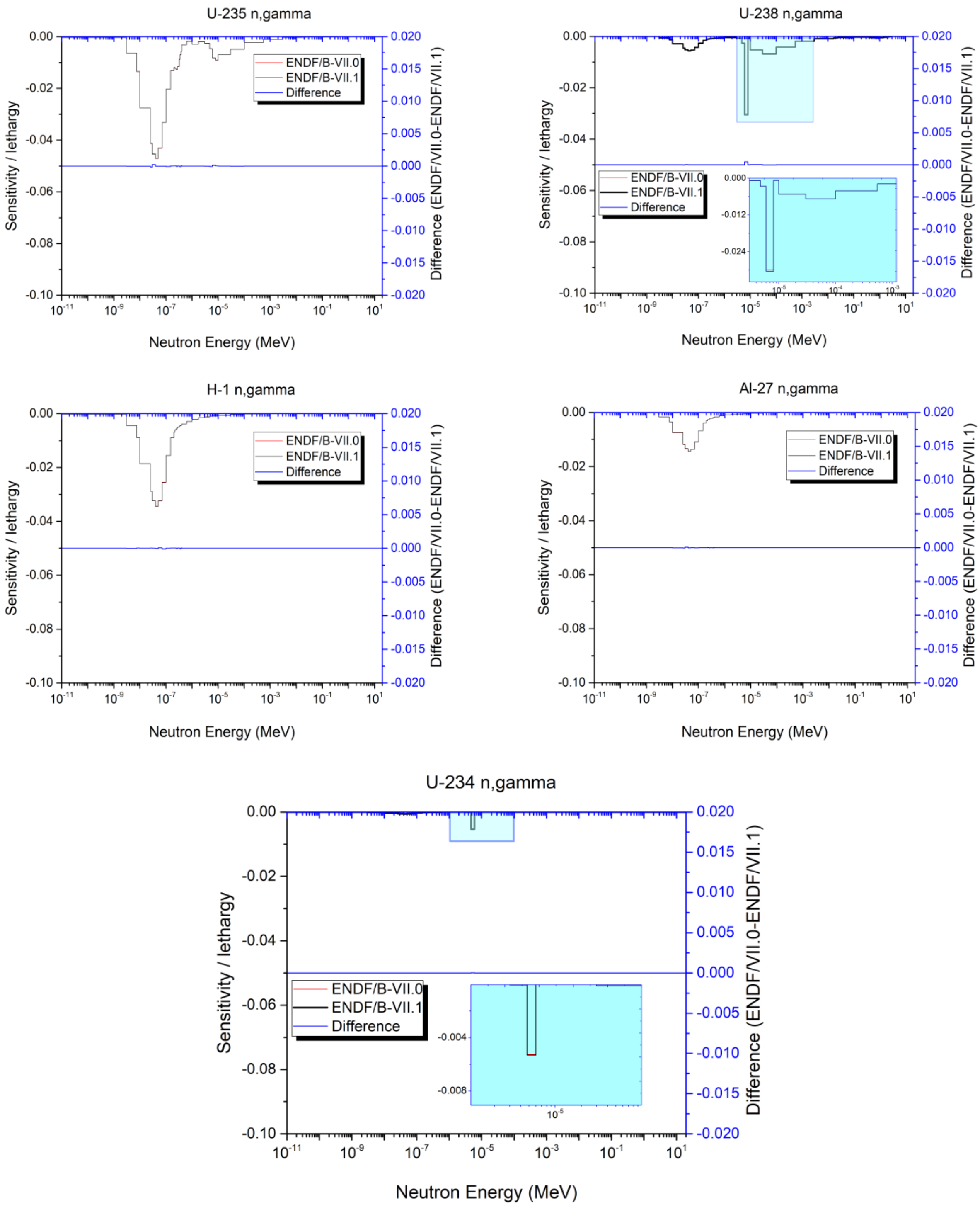

Figure 11. Energy dependent sensitivities of k_{inf} of LEU fuel assembly for ENDF/B-VII.0 and ENDF/B-VII.1 (negative contribution)

Figure 12 depicts the uncertainty of the U-235, Al-27, H-1, O-16, U-234 and U-238 isotopes in the LEU fuel obtained by Whisper1.1 program. The uncertainty caused by cross section error of ENDF/B-VII.0 for LEU fuel assembly is about the same with the one of HEU fuel assembly. The uncertainty of the reaction cross-section of the U-235 isotope is 0.386% of reactivity effect while the total ν has the greatest uncertainty due to the contribution of each individual reaction is 0.297%, followed by the neutron capture reaction (0.199%) and fission (0.088%). The cross-sectional uncertainty of all reactions of Al-27 is 0.154% mostly from the neutron capture reaction. The uncertainty from the reaction cross-sections of H-1, O-16, U-234 and U-238 is 0.044%, 0.006% 0.013% and 0.070%, respectively.

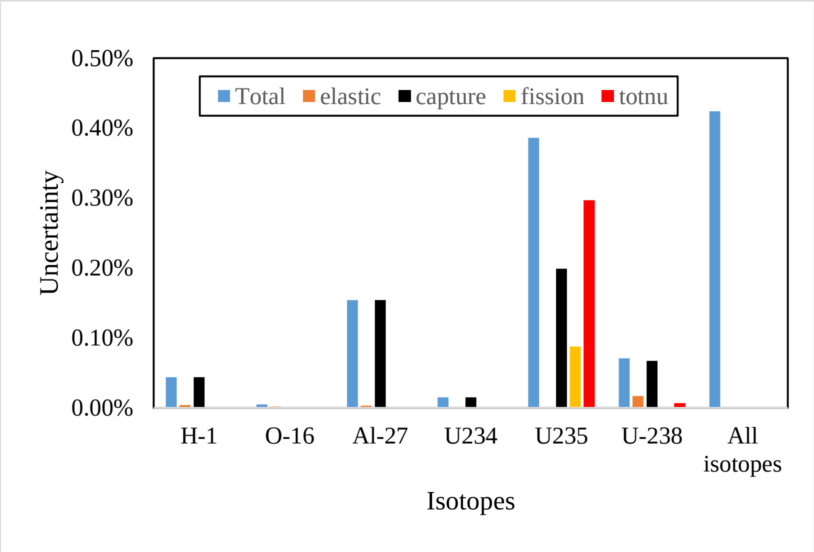

Figure 12. Uncertainty of k_{inf} caused by ENDF/B-VII.0 library for LEU fuel assembly.

5. Conclusion and Future Work

Sensitivity and uncertainty analysis has been performed for the k_{inf} of HEU and LEU fuel assemblies of the DNRR using MCNP6.1 and ENDF/B-VII.0 and ENDF/B-VII.1 nuclear data libraries. The most significant sensitivity coefficients in positive contribution are the coefficient total ν and fission reaction of U-235, elastic scattering reaction of H-1, inelastic scattering of thermal neutron, total ν of U-238 and in negative contribution are capture reactions of U-235, H-1, Al-27, U-238 and U-234 isotopes. The dependence of the sensitivity on the incident neutron energies of the isotopes has also been analyzed to provide better knowledge of which energy region the cross section should be adjusted in order to receive a more reliable calculation results. The quite large discrepancy between sensitivities of elastic scattering reaction cross section of H-1 and inelastic scattering of thermal neutron of two libraries are found because of the change in neutron spectra of HEU and LEU fuel assemblies using the two library versions. With the 44 energy group sensitivity profiles obtained from MCNP6.1, Whisper1.1 module was used to estimate the uncertainty of the isotopes from the ENDF/B-VII.0 library using 44-group covariance matrix. The uncertainty of k_{inf} of the HEU and LEU fuel assembly caused by all the isotope cross section errors is quite significant (0.447% and 0.424% of reactivity effect, respectively). Among isotopes, the uncertainty of U-235 is the greatest at about 0.4%, followed by Al-27, H-1, U-238 and O-16.

In future work, we plan to conduct the sensitivity and uncertainty analysis for the new nuclear data libraries such as ENDF/B-VIII.0, JENDL4.0 for the DNRR whole core configuration in order to investigate the impact of nuclear data error on the research reactor calculation results.

Acknowledgments

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