# EG0700571
# The Effect of Uncertainty in Cross Sections in the ENDF/B-VI Library on the Neutronic Parameters of the ETRR-2 Reactor.

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## ABSTRACT

**The elimination of a large number of approximations that lead to numerous errors in the neutronic reactor calculations was the main purpose behind developing Monte Carlo codes. The MCNP series of codes (Monte Carlo Nuclear Particle) are developed and extensively used in neutronic core calculations. The neutronic data input to these codes, the pointwise cross section files as presented by ENDF libraries or similar ones, are comprehensive and detailed. However the major sources of errors in the core calculations stem from uncertainties in the cross section data. In this paper the effect of estimated uncertainty in the values of cross sections in the ENDF/B-VI library, on the neutronic parameters of the ETRR-2 reactor is studied. MCNP code is used to simulate a three-dimensional model for the reactor core considering all the materials composition and geometrical details. Perturbation technique is used to determine the effect of uncertainties in cross sections for a number of isotopes in the reactor on the fission rates and a comparison is made between the fission rate values with and without the uncertainty values for the different cross section types and different energy ranges. It is shown that for all the considered isotopes the effect of uncertainties in the cross section data on the fission rate values is small, where the differences in fission rates do not exceed 10% and this value is accepted. The perturbations in nuclear data for high-energy ranges are estimated to have no effect on the fission rate values.**

***Key words: Uncertainty in Cross Section Data/ ENDF/B-VI/ ETRR-2.***

## INTRODUCTION

Uncertainties in the cross section data are considered to be the most important source of errors in core calculations and it predominates over other sources of uncertainties. So it is important to collect the uncertainties in cross section values obtained for each of the isotopes used in the ETRR-2 reactor and study their effect on the calculations of the reactor.

The Cross-Section Evaluation Working Group (CSEWG) Standards Committee supplied a set of expanded uncertainty estimates for the standard cross section reactions, which are available for some of the isotopes in the tapes of the ENDF/B-VI (Release 8) library (1). If any new experiment is performed on a given standard using the best techniques, approximately 2/3 of the results should fall within these expanded uncertainties. It was found that in the ENDF/B-VI library, the uncertainty estimates values are available for the H-1, U-235 and C-natural isotopes, which are used in the ETRR-2 reactor. Table (1) includes the values of the estimated uncertainties percent for each

of the three isotopes, specified over given energy ranges (Kev) and for certain type of cross section.

**Table (1): Estimated uncertainty values in the ENDF/B-VI library**

| Isotope | Energy Range (Kev) | Estimated Uncertainty (%) | Type of cross section |
|---|---|---|---|
| H-1 | 1.0E-08 ---- 20000 | 0.2 | Total cross section |
| U-235 | 2.53E-05 | 0.2 | Fission cross section |
| | 150 ---- 600 | 1.5 | |
| | 600 ---- 1000 | 1.6 | |
| | 1000 ---- 3000 | 1.8 | |
| | 3000 ---- 6000 | 2.3 | |
| | 6000 ---- 10000 | 2.2 | |
| | 1000 ---- 12000 | 1.8 | |
| | 12000 ---- 14000 | 1.2 | |
| | 14000 ---- 14500 | 0.8 | |
| | 14500 ---- 15000 | 1.5 | |
| | 15000 ---- 16000 | 2 | |
| | 16000 ---- 17000 | 2.5 | |
| | 17000 ---- 19000 | 3 | |
| | 19000 ---- 20000 | 4 | |
| C-nat | 1 ---- 500 | 0.46 | Elastic scattering cross section |
| | 500 ---- 1500 | 0.53 | |
| | 1500 ---- 1800 | 0.60 | |

The estimated uncertainty values for the other isotopes found in the core of the reactor are not calculated yet in the ENDF/B-VI tapes, so one had to find them in other sources. For instant S. Tagesen [(2)] et al. have evaluated all the important integral neutron cross sections for neutron energies up to 20 MeV using the code GLUCS. From these evaluations they have determined the estimated uncertainty value for the total cross section of Be-9 isotope to be 2% at incident neutron energy equal to 14 MeV.

For another set of isotopes, the estimated uncertainty values can be taken from the NEA High Priority Nuclear Data Request List (HPRL) (3). HPRL is a compilation of the highest priority nuclear data requirements, to provide a guide for nuclear data improvements to those planning measurement, nuclear theory and evaluation programs. Since uncertainty is defined to be the limits of accuracy, thus the degree of accuracy values in percent for the Ni-58, Si-natural, O-16, Fe-56, Cr-52 and U-238 isotopes can be taken from the HPRL status report. They are specified over given energy ranges (Kev) and for certain type of cross section as seen in table (2).

**Table (2): Required accuracy values presented in the HPRL report**

| Isotope | Energy Range (Kev) | Required Accuracy (%) | Type of cross section |
|---|---|---|---|
| Ni-58 | 6000----10000 | 10 | Absorption cross section |
| Si-nat | 9000----13000 | 10 | Absorbtion cross section |
| O-16 | 1000 ---- 14000 | 5 | Absorbtion cross section |
| Fe-56 | 2000 ---- 5000 | 10 | Elastic Scattering cross section |
| Cr-52 | 0.01 ---- 20000 | 3 | Total cross section |
| U-238 | $E_{th}$ ---- 20000 | 1 | Fission cross section |

The MCNP4B code (4) is used to study the effect of the collected values of uncertainties on the fission rate values. MCNP4B is a general purpose Monte Carlo N-particle transport code developed to simulate neutron, photon and electron transport and it can use pointwise cross section data as input. This study is applied to the Egypt second research reactor (ETRR-2) reactor (5).

## THE ETRR-2 DESCRIPTION

ETRR-2 reactor is a multipurpose, heterogeneous, open pool and material testing reactor. It went critical for the first time on 27th November 1997 and it can be operated up to a maximum power of 22 MW with 19.7 % fuel enrichment. It uses light water as a coolant and moderator, and beryllium as a reflector. The first operational core was identified as core configuration (2/98) and it was operated for 0.25 day, then it is followed by core (3/98) (6) that was assembled on 9th of March 1998 and operated until 7.3 days.

A horizontal view for the core of ETRR-2 is presented in figure (1), the core is an array of fuel elements, reflectors, absorber rods, gadolinium injection chambers and irradiation devices. It consists of 6 x 5 grid positions to hold 29 square fuel elements and one grid space for cobalt 60 production. There are two zones dividing the grid to hold six guide boxes for control rod insertion. Around the core there are four gadolinium injection chambers, beryllium reflectors and irradiation boxes.

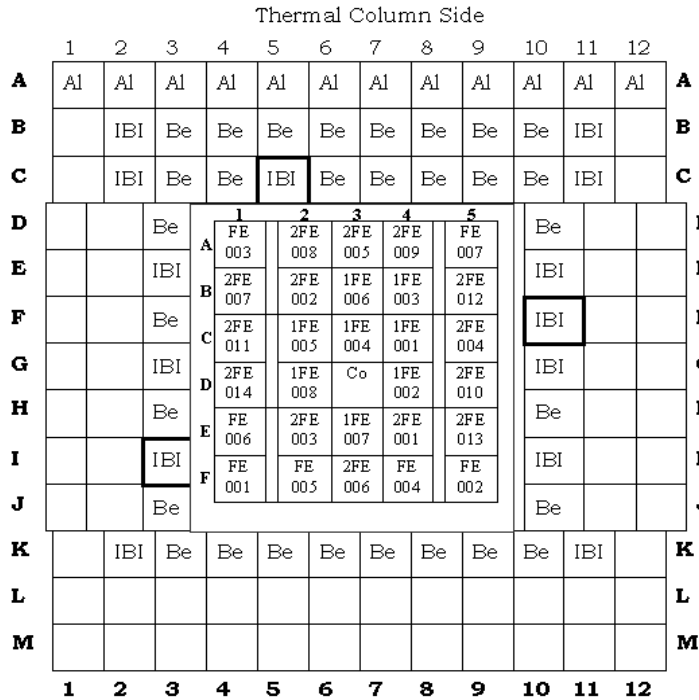


**Figure (1): Horizontal view of core 3/98**

## CACULATIONS AND RESULTS

After collecting the values of both the estimated uncertainties and the degrees of accuracy in cross section data for most of the isotopes used in ETRR-2 reactor, their effects on the neutronic parameters of the ETRR-2 reactor are studied. A three-dimensional model is simulated to represent the core configuration 3/98 of the reactor using MCNP4B code and the fission rate values are determined without considering the uncertainty values. Then by using the perturbation technique, in addition to the values collected for both the estimated uncertainty and degree of accuracy, new fission rate values are obtained for each of the previous mentioned isotopes.

The perturbation card in MCNP4B code can be used to represent perturbations in cell material density, composition, or reaction cross-section data. Calculations are performed twice for each isotope, once with the addition of the uncertainty value and the other with subtracting it. Finally for each isotope a comparison is made between the fission rate values with and without uncertainty values for the different cross section types and different energy ranges. From these comparisons one can determine the effect of uncertainties in the basic cross section data on calculations.

Tables (3) to (8) provide the differences between the fission rate values (ΔRF%) obtained with and without uncertainty values for each of the collected isotopes, at different energy ranges and for different cross section types. Table (3) presents the fission rate differences for the isotopes H-1, Ni-58, Be-9 and Si-natural. Each of the previous isotopes is found in a single region in the reactor core and have a single uncertainty value, which is either added or subtracted from the original cross section value.

It is shown that for the hydrogen and nickel isotopes the effect of uncertainties in the cross section data on the fission rate values is very small. But for the beryllium and silicon isotopes the difference in fission rate values is equal to zero because the energy values specified for each of them are large and it is known that fast neutrons do not make fissions at high energy ranges. The table also includes three ranges of uncertainty values for C-natural specified over three different energy ranges but for the same cross section type that is the elastic cross section. The carbon isotope exists in only one region inside the core that is the neutrons beam region, which has no effect on the fission rate values and hence the difference will be equal to zero.

**Table (3): The fission rate differences for H-1, Ni-58, Be-9, Si and C-natural.**

| Isotope | Energy Range (Mev) | Estimated Uncertainty (%) | Type of cross section | ΔRF% |
|---|---|---|---|---|
| H-1 | 1.0E-11→ 20 | -0.2 --- 0.2 | Total cross section | 0.097 --- -0.096 |
| Ni-58 | 6 → 10 | -10 --- 10 | Absorption cross section | -8.69E-04 --- 0.03 |
| Be-9 | 14 | 2 | Total cross section | Zero |
| Si-nat | 9 → 13 | 10 | Absorption cross section | Zero |
| C-nat | 1.0E-03→0.5 | -0.46 --- 0.46 | Elastic cross section | Zero |
| C-nat | 0.5 → 1.5 | -0.53 --- 0.53 | Elastic cross section | Zero |
| C-nat | 1.5 → 1.8 | -0.6 --- 0.6 | Elastic cross section | Zero |

Table (4) presents the fission rate differences obtained due to uncertainties in fission cross section of U-235 specified over energy ranges from 2.53E-08 to 20 Mev, these ranges are divided into a number of intervals. U-235 is found in each of the three types of fuel elements, so the fission rate values are calculated for each of them. Calculations are performed twice one with positive uncertainty value and the other with negative value. It is shown that the differences in the calculated fission rates are very small for energy ranges below 14 Mev, and have zero values for energy ranges above 14 Mev.

**Table (4): The fission rate differences for U-235.**

| Energy Range (Mev) | Estimated Uncertainty (%) | ΔRF% (FE-404) | ΔRF% (FE-209) | ΔRF% (FE-148) |
|---|---|---|---|---|
| 2.53E-08 | 0.2 | -3.07E-4 | -6.35E-4 | -5.04E-4 |
| 2.53E-08 | -0.2 | 3.1E-4 | 6.34E-4 | 5.03E-4 |
| 0.15 → 0.6 | 1.5 | -2.51E-3 | -3.03E-3 | -1.63E-3 |
| 0.15 → 0.6 | -1.5 | 2.46E-3 | 2.96E-3 | 1.59E-3 |
| 0.6 → 1 | 1.6 | -1.51E-3 | -1.71E-3 | -9.65E-4 |
| 0.6 → 1 | -1.6 | 1.48E-3 | 1.67E-3 | 9.4E-4 |
| 1 → 3 | 1.8 | -4.17E-3 | -4.92E-3 | -2.73E-3 |
| 1 → 3 | -1.8 | 4.05E-3 | 4.78E-3 | 2.66E-3 |
| 3 → 6 | 2.3 | -1.45E-3 | -1.7E-3 | -9.57E-4 |
| 3 → 6 | -2.3 | 1.4E-3 | 1.64E-3 | 9.21E-4 |
| 6 → 10 | 2.2 | -2.38E-4 | -2.56E-4 | -1.49E-4 |
| 6 → 10 | -2.2 | 2.28E-4 | 2.47E-4 | 1.44E-4 |
| 10 →12 | 1.8 | -5.61E-6 | -8.7E-6 | -2.24E-6 |
| 10 → 12 | -1.8 | 5.43E-6 | 8.45E-6 | 2.18E-6 |
| 12 → 14 | 1.2 | -3.59E-6 | -1.65E-6 | -1.01E-6 |
| 12 → 14 | -1.2 | 3.51E-6 | 1.61E-6 | 9.88E-7 |
| 14 →14.5 | 0.8 | Zero | Zero | Zero |
| 14.5 → 15 | 1.5 | Zero | Zero | Zero |
| 15 → 16 | 2 | Zero | Zero | Zero |
| 16 → 17 | 2.5 | Zero | Zero | Zero |
| 17 → 19 | 3 | Zero | Zero | Zero |
| 19 → 20 | 4 | Zero | Zero | Zero |

Table (5) presents the fission rate differences obtained due to uncertainties in absorption cross section of O-16 specified over energy range from 1 to 14 Mev. Oxygen isotope is found in the three types of fuel elements, chimney, coolant and neutron beam regions. For each region, calculations of the fission rate values are performed twice one with negative uncertainty value (-5%) and the other with positive value (5%). It is shown that the differences in the calculated fission rates are very small for the fuel elements and coolant, and it is zero for other regions.

**Table (5): The fission rate differences for O-16.**

| Cell | ΔRF % |
|---|---|
| FE-404 | -5.79E-03 → 6.08E-03 |
| FE-209 | -8.69E-03 → 8.69E-03 |
| FE-148 | -5.79E-03 → 5.79E-03 |
| Chimney | Zero |
| Coolant | -0.01 → 0.01 |
| Beam Cells | Zero |

Similarly table (6) presents the fission rate differences obtained due to uncertainties in elastic scattering cross section of Fe-56 specified over energy range from 2 to 5 Mev. The iron isotope is found in the regions of chimney and control rods. For each region calculations of the fission rate values are performed twice one with negative uncertainty value (-10%) and the other with positive value (10%). It is shown that the differences in the calculated fission rates are also very small.

**Table (6): The fission rate differences for Fe-56.**

| Cell | ΔRF % |
|---|---|
| Chimney | 3.48E-03 → -3.48E-3 |
| Control Rods | 0.16 → 0.43 |

Table (7) presents the fission rate differences obtained due to uncertainties in total cross section of Cr-52 specified over energy range from 0.01 to 20 Mev. The isotope is found in the regions of chimney, cobalt, clad, neutron beams, control rods and their guide boxes. For each region calculations of the fission rate values are performed twice one with negative uncertainty value (-3%) and the other with positive value (3%). It is shown that the differences in the calculated fission rates are very small.

**Table (7): The fission rate differences for Cr-52.**

| Cell | ΔRF % |
|---|---|
| Clad, cobalt, guide box, beam cells | 2.61E-03 → -2.61E-03 |
| Chimney | 3.48E-03 → -3.48E-03 |

| | |
|---|---|
| Control rods | 0.03 → 0.01 |

Finally table (8) presents the fission rate differences obtained due to uncertainties in fission cross section of U-238 specified over energy range from the threshold energy to 20 Mev. The isotope is found in the three types of fuel elements. For each type calculations of the fission rate values are performed twice one with negative uncertainty value (-1%) and the other with positive value (1%). It is shown that the differences in fission rates do not exceed ±10% and this value is accepted.

**Table (8): The fission rate differences for U-238.**

| **Cell** | **ΔRF** % |
|---|---|
| FE-404 | 4.95 → -2.84 |
| FE-209 | 9.26 → -6.73 |
| FE-148 | 6.29 → -4.91 |

It can be observed that for all the considered isotopes the effect of uncertainties in the cross section data on the fission rate values is very small, where the differences in fission rates do not exceed 10% and this value is accepted. Also the perturbations in nuclear data for high-energy ranges are estimated to have no effect on the fission rate values.

## CONCLUSION

It is shown that the effect of uncertainties in the cross section data for most of the isotopes of the ETRR-2 reactor, on the fission rate values is small, where the differences in fission rates do not exceed 10% and this value is accepted. Also the perturbations in nuclear data for high-energy ranges are estimated to have no effect on the fission rate values.

## REFERENCES

(1) Dermott E. Cullen, "Point 2004 A Temperature Dependent ENDF/B-VI, Release 8 Cross Section Library", UCRL-TR-202284, U.S. Department of Energy, Lawrence Livermore National Laboratory (April 1, 2004).

(2) S. Tagesen, H. Vonach and V.G. Pronyaev, "Evaluation of all important neutron cross sections for $^{9}$Be and of the shape of the angular-energy distribution of secondary neutrons", (July 1997).

(3) NEA Nuclear Science Committee working party on International Evaluation Co-operation subgroup C, "The NEA High Priority Nuclear Data Request List" (March 2001).

(4) Judith F. Briesmeister, "MCNP—A general Monte Carlo N—Particle Transport Code, Version 4B", LA-12625-M, Version 4B, Manual (March 1997).

(5) NCNSR, Atomic Energy Authority, "Safety Analysis Report for ETRR-2 (SAR)" (June 2003).

(6) Atomic Energy Authority, "Measurements on the proposed start up core", protocol No. 767-COMM-053 (March 1998).