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RESONANCE SELF-SHIELDING CALCULATIONS USING THE PROBABILITY
TABLE METHOD

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August 2, 1976

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RESONANCE SELF-SHIELDING CALCULATIONS USING THE PROBABILITY TABLE METHOD

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Abstract

Resonance self-shielding calculations are presented for fissile material (U-235) and iron using the Probability-Table-Method in a Monte Carlo code.

Introduction

A multigroup-Monte-Carlo neutron transport code that uses the Probability-Table-Method⁽¹⁻⁴⁾ for all neutron energies has been developed. This code was developed from the TARTNP⁽⁵⁾ multigroup-Monte-Carlo code. The reason for developing the code was to automatically include self-shielding effects and self-consistent reaction rate estimation⁽⁶⁾ for the elastic, capture and fission reactions for all elements or isotopes in a calculation.

The Probability-Table-Method code has the option to do calculations as a standard-multigroup code. This option allows for direct comparison between the two methods

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(i.e., standard-multigroup or multigroup with cross section probability tables). The Probability-Table-Method code uses 175 energy groups from 0.019 eV to 20.0 MeV.

Resonance self-shielding is important in some calculations for energies less than a few kilovolts in high-atomic-number elements. The self-shielding phenomenon is also important in the million-electron-volt range for intermediate-atomic-number elements.⁽¹⁾ The variation of resonance self-shielding with energy and atomic number made it necessary to construct a general method of solution for this problem.

The results of calculations reported here are for a neutron cross section library that has two cross section sets, (i.e., bands⁽⁴⁾) for the total, elastic, capture (n, γ) and fission for each energy group. The code has been written, however, to handle an arbitrary number of cross section bands⁽⁴⁾ per energy group. The method and assumptions required to prepare the cross section library are given in Refs. 4 and 5.

Discussion of Calculations

Calculations were done for two different problems. The first was the Bramblett-Czirr highly-enriched U²³⁵ fission plate experiment⁽⁷⁾ where the fissions caused by the uncollided flux were measured as a function of thickness

of U^{235} for a number of different energy ranges. The calculations were done only for energies less than 100 eV.⁽⁸⁾ The calculations for the highly-enriched U^{235} plate typify the resonance self-shielding in reactor core material. The second problem was a 30 cm thick infinite slab of iron that was chosen to be representative of shielding material. The resonance structure in iron extends from about 1.0 keV to about 3.0 MeV.⁽⁸⁾

U^{235} Fission Plate Problem

A sketch of the calculational model used for the fission plate problem is given below:

Table 1 presents the number of energy groups used by each code for the several energy ranges for which the fission plate calculations were done.

Calculations for the fission plate problem were made with a 2020-group-Monte-Carlo code and the Probability-Table-Method code in both modes. The exact calculation of the fissions for a fission plate of thickness X was determined by Eq. (1):

$$F(E) = \frac{\int_{E_g}^{E_{g+1}} \frac{\sigma_f(E)}{E} e^{-\Sigma_T(E)X} dE}{\int_{E_g}^{E_{g+1}} \frac{\sigma_f(E)}{E} dE} \quad (1)$$

Figure 1 presents the results of the Probability-Table-Method, exact, and 2020-group-Monte-Carlo calculations for a hypothetical U^{235} fission plate. Because of the excellent agreement between the exact and 2020-group calculations for U^{235} it is assumed that the two methods of calculation would agree for the isotopic mixture in the fission-plate experiment. The Probability-Table-Method is in reasonable agreement with the other two methods, but appears to have over-emphasized the larger cross sections.

Figures 2 through 8 give the results of the calculation for the fission plates. In general, the agreement is reasonably good between calculation by the Probability-Table-Method, the 2020-group-Monte-Carlo code, and experiments, but the 175-group calculation deviates by as much as 100% from the

other calculations and the experiments. (The Probability-Table-Method calculation deviates by amounts up to 16% from the 2020-group calculation.) Figures 2 and 8 show large differences between calculations and experiment.

The calculated results are in agreement in Fig. 2. The one-group total cross section that can be inferred from the experimental data for the energy range from 0.464 to 1.0 eV is 12% smaller than the one-group cross section that can be inferred from the calculation. The measured total cross sections in this range have experimental errors of only 2 to 3 percent so the 12% deviation is not understood. The Appendix contains plots of the experimental and evaluated cross section data.

In Fig. 8 the experimental and calculated results are not in agreement because the resonance structure in U^{235} does not extend above 82 eV in the evaluated data.⁽⁷⁾

Iron Problem

A sketch of the iron problem is given below.

The source was a

$\frac{1}{E}$ Spectrum

from 1.0 keV to 20.0 MeV

The exact uncollided transmission through 30.0 cm of iron was calculated from Eq. (2):

$$T(X) = \int_{1.0 \text{ keV}}^{20 \text{ MeV}} e^{-\Sigma_T(E)X} \frac{dE}{E} \quad (2)$$

The 2020-group-Monte-Carlo code and the Probability-Table-Method code in both modes were used to calculate the uncollided and total transmission in the iron as a function of depth. The iron problem used a different group structure than was used for the fission problem. There were 1980 of the 2020 groups between 1.0 keV and 20.0 MeV for the iron calculation while the Probability-Table-Method code had 85 groups from 1.0 keV to 20.0 MeV.

In the transmission calculation through 30.0 cm of iron, self-shielding is important as is demonstrated by the results of the calculations which are presented in Table 2. The 2020-group and Probability-Table-Method calculations of the uncollided flux agree reasonably well with the exact calculation while; the standard-175-group calculation is too small by an order of magnitude. The total flux calculated by the 175-group-Monte-Carlo code is 14.5% less than that calculated by either the 2020-group-Monte-Carlo code or the Probability-Table-Method code. Time comparisons show that the Probability-Table-Method code required 20%

more computer-time than either the 2020-group or 175-group codes. Since the answers are the same for the Probability-Table-Method code and the 2020-group code, the trade-off is in time used vs. memory-space required.

Conclusions

1. The 175-group-Probability-Table-Method, with two bands, is superior to the standard-175-group-Monte-Carlo method calculating self-shielding for fissile materials.

2. Calculations using the Probability-Table-Method agreed to within 16% with calculations using the 2020-group-Monte-Carlo method for the fission-plate problems. The standard-175-group calculation deviated as much as 100% from the 2020-group calculation.

3. The Probability-Table-Method calculation of the uncollided flux through 30 cm of iron was in reasonable agreement with the exact calculation, and the calculated total flux was in excellent agreement with the 2020-group-Monte-Carlo calculation. The standard-175-group-Monte-Carlo calculation differs from the exact calculation by a factor of 10 for the uncollided transmission.

4. To calculate both problems to the accuracy presented with a single group structure would require more than 3000 groups in a multigroup-Monte-Carlo code. Including the unresolved resonance region would increase the

number of groups by an additional one- or two-orders of magnitude.

5. The Probability-Table-Method provides a mechanism for avoiding excessively large cross section files without sacrificing accuracy or causing a significant increase in computer-time.

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Table 1. Number of groups in fission problem.

<u>Energy Range</u>	<u>PTM and Multigroup Code</u>	<u>2020 Group Code</u>
.464 - 1.0 eV	2	4
1.0 - 2.15 eV	3	11
2.15 - 4.64	2	26
4.64 - 10.0	5	55
10.0 - 21.5	7	116
21.5 - 46.4	14	251
46.4 - 100.	22	399

Table 2. Calculated transmission through 30 cm of iron.

METHOD	UNCOLLIDED FLUX	(TOTAL MINUS UNCOLLIDED) FLUX
Theoretical	.00478	
Probability Table Method	.00518 ±.00015	.1518 ±.0008
2020 Group Monte Carlo	.00430 ±.00016	.1521 ±.0007
175 Group Monte Carlo	<u>.000490</u> ±.000050	.1300 ±.0013

APPENDIX

U^{235} CROSS SECTIONS

07/04/76

Total
Cross Section

92-U -235
ENDL-7168

07/04/76

Total
Cross Section

92-U -235
ENDL-7168

MeV

11-2

92-U -235 -2

Barns

07/04/76

Total
Cross Section

92-U -235

ENDL-7168

Barns

07/04/76

Total
Cross Section

92-U -235
ENDL-7168

1.1-4

MeV

92-U -235 -4

Barns

07/04/76

Total
Cross Section

92-U -235

ENDL-7168

07/04/76

Total
Cross Section

92-U -235

ENDL-7168

11-6

92-U -235 -6

Barns

07/04/76

Total
Cross Section

92-U -235
ENDL-7168

11-7

92-U -235 -7

07/04/76

Total
Cross Section

92-U -235
ENDL-7168

07/04/76

Elastic Cross Section

92-U -235
ENDL-7168

Barns

07/04/76

Elastic Cross Section

92-U -235
ENDL-7168

11-10

92-U -235 -10

07/04/76

Elastic Cross Section

92-U -235

ENDL-7168

Barns

07/04/76

Fission Cross Section

92-U -235

ENDL-7168

Barns

11-11

92-U -235 -11

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

11-12

92-U -235 -12

Barns

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

11-13

92-U -235 -13

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

11-14

92-U -235 -14

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

11-15

MeV

92-U -235 -15

Barns

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

11-16

92-U -235 -16

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

11-17

92-U -235 -17

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

Barns

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

11-19

92-U -235 -19

Barns

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

11-20

92-U -235 -20

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

Barns

-04-

11-21

92-U -235 -21

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

Barns

-14-

11-22

92-U -235 -22

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

Barns

11-23

92-U -235 -23

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

11-24

92-U -235 -24

Barns

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

(Expanded Below)
ECSIL-1017 (60)

ECSIL-1017 (60)

ECSIL-1017 (60)

Barns

-44-

11-25

Mev

92-U -235 -25

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

Barns

-54-

11-26

Mev

92-U -235 -26

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

Barns

-94-

11-27

92-U -235 -27

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

Barns

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

Barns

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

07/04/76

Fission
Cross Section

92-U -235
ENDL-7168

11-31

92-U -235 -31

07/04/76

Fission
Cross Section

92-U -235

ENDL-7168

07/04/76

Fission
Cross Section

92-U -235

11-32

92-U -235 -32

07/04/76

(n,γ)
Cross Section

92-U -235

ENDL-7168

07/04/76

(n,γ)
Cross Section

92-U -235
ENDL-7168

11-34

Mev

92-U -235 -34

07/04/76

(n,γ)
Cross Section

92-U -235
ENDL-7168

Mev

11-35

92-U -235 -35

Barns

07/04/76

(n,γ)
Cross Section

92-U -235

ENDL-7168

(Expanded Below)
ECSIL-3262 (72)

ECSIL-3262 (72)

ECSIL-3262 (72)

11-36

Mev

92-U -235 -36

Barns

07/04/76

(n,γ)
Cross Section

92-U -235
ENDL-7168

07/04/76

Prompt Neutrons/Fission

92-U -235
ENDL-7168

11-37

92-U -235 -37

07/04/76

Prompt Neutrons/Fission

92-U -235

ENDL-7168

07/04/76

Neutrons/Nonelastic Event

92-U -235

11-38

92-U -235 -38

07/04/76

Neutrons/Nonelastic Event

92-U -235

11-39

MeV

92-U -235 -39

Ratio

07/04/76

Neutrons/Nonelastic Event

92-U -235

-59-

11-40

92-U -235 -40

07/04/76

Capture/Fission Ratio

92-U -235

11-41

92-U -235 -41

07/04/76

Capture/Fission Ratio

92-U -235

Ratio

07/04/76

Total
Cross Section

92-U -235

REFERENCE	ENERGY MEV	DELTA MEV	X-SEC BARNs	DELTA BARNs	REFERENCE	ENERGY MEV	DELTA MEV	X-SEC BARNs	DELTA BARNs
61 1332	4.6700- 7	4.4200- 8	1.1570+ 2	5.8000+ 0	55 902	5.8300- 7		1.0600+ 2	
55 902	4.8000- 7		1.2300+ 2		58 1463	9.9300- 7		8.6000+ 1	
61 1332	5.6100- 7	5.8196- 8	1.0010+ 2	5.8000+ 0					

07/04/76

Fission
Cross Section

92-U -235

REFERENCE	ENERGY MEV	DELTA MEV	X-SEC BARNs	DELTA BARNs	REFERENCE	ENERGY MEV	DELTA MEV	X-SEC BARNs	DELTA BARNs
58 1021	4.9200- 7		8.1300+ 1	2.5000+ 0	58 1021	7.4400- 7		5.5400+ 1	
58 1021	5.9900- 7		6.1100+ 1		58 1021	9.4900- 7		6.3700+ 1	

07/04/76

Neutrons/Nonelastic Event

92-U -235

REFERENCE	ENERGY MEV	DELTA MEV	RATIO RELATIVE	DELTA RELATIVE	REFERENCE	ENERGY MEV	DELTA MEV	RATIO RELATIVE	DELTA RELATIVE
55 1212	4.8000- 7	5.0000- 8	8.5600- 1	2.0000- 2	55 1212	8.4000- 7	1.0000- 7	8.2600- 1	4.0000- 2
55 1212	6.2000- 7	7.0000- 8	8.2600- 1	4.0000- 2	55 1212	1.2000- 6	2.0000- 7	8.2600- 1	4.0000- 2

07/04/76

Capture/Fission Ratio

92-U -235

REFERENCE	ENERGY MEV	DELTA MEV	RATIO	DELTA
58 1021	4.9200- 7		1.8800- 1	

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