# Lumped Group Constants of FP Nuclides for Fast Reactor Shielding Calculation Based on JENDL-3.2

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## ABSTRACT

The lumped group constants of FP nuclides used for a fast reactor shielding calculation were computed based on the JENDL-3.2 nuclear data library and were compiled to the JSDJ2/ JFTJ2 set. The effect of FP nuclides on neutron flux calculations was evaluated in the JOYO experimental fast reactor. These tests showed that conventional calculations that ignored FP overestimated neutron flux by about 2%.

**Keywords :** lumped group constants, FP, fast reactor, JOYO, shielding, JENDL-3.2

## 1. Introduction

Fission products (FPs) were not considered in conventional fast reactor shielding analyses that were predominantly developed in clean core experiments like the JASPER program [1, 2]. However, in power reactors with high burn-up, the accumulation of FP affects the neutron balance so it cannot be neglected in the neutron flux calculation. In this study, the group constants of FP nuclides were computed based on the JENDL-3.2 [3] nuclear data library and these were compiled to the JSDJ2 [4] set. Using the constants, the effect of the FP nuclides on shielding calculation was evaluated in the JOYO experimental fast reactor.

## 2. Lumped FP Group Constants

### 2.1 Selection of FP Nuclides

The flow chart for computing the lumped FP constants is illustrated in Fig. 1.

Initially, FP nuclides need to be considered for the group constants. Generation and depletion for nearly 880 FP nuclides can be computed with the ORIGEN2 [5] burn-up calculation. The calculation uses the specification and material contents of the JOYO Mk-II driver as an example of fast reactor MOX fuel.

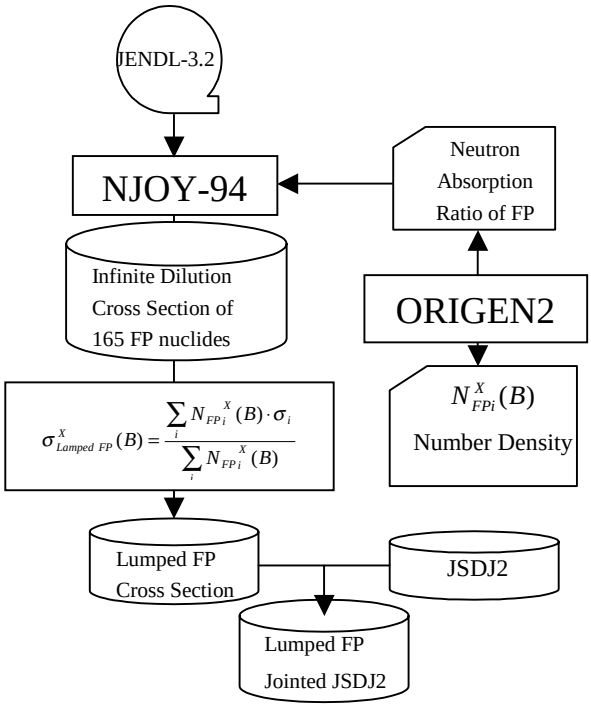


Fig.1 Flow Chart of Computing FP Group Constant

Table.1 The neutron absorption ratio of FP nuclides

| | Nuclide | R | A | | Nuclide | R | A | | Nuclide | R | A | | Nuclide | R | A |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | Ru-101 | 8.93% | 8.93% | 54 | Ce-144 | 0.18% | 96.56% | 107 | Eu-151 | 0.01% | 99.68% | 160 | Ag-107 | 0.00% | 99.81% |
| 2 | Pd-105 | 7.90% | 16.83% | 55 | Gd-157 | 0.18% | 96.73% | 108 | Cd-116 | 0.01% | 99.69% | 161 | Be-9 | 0.00% | 99.81% |
| 3 | Tc-99 | 7.06% | 23.89% | 56 | Pd-110 | 0.14% | 96.87% | 109 | Sn-118 | 0.01% | 99.70% | 162 | Nb-93 | 0.00% | 99.81% |
| 4 | Rh-103 | 6.02% | 29.91% | 57 | Mo-99 | 0.14% | 97.01% | 110 | Cd-110 | 0.01% | 99.71% | 163 | Ga-71 | 0.00% | 99.81% |
| 5 | Cs-133 | 5.72% | 35.63% | 58 | Gd-156 | 0.13% | 97.14% | 111 | Se-78 | 0.01% | 99.72% | 164 | Li-7 | 0.00% | 99.81% |
| 6 | Pd-107 | 4.65% | 40.29% | 59 | Cd-113 | 0.11% | 97.25% | 112 | Xe-130 | 0.01% | 99.72% | 165 | Ga-69 | 0.00% | 99.81% |
| 7 | Mo-97 | 4.54% | 44.83% | 60 | Cs-134 | 0.11% | 97.35% | 113 | Ba-137 | 0.01% | 99.73% | | | | |
| 8 | Sm-149 | 4.39% | 49.21% | 61 | Eu-154 | 0.10% | 97.46% | 114 | Gd-160 | 0.01% | 99.74% | | | | |
| 9 | Pm-147 | 3.77% | 52.98% | 62 | Ce-140 | 0.10% | 97.56% | 115 | Ba-140 | 0.01% | 99.74% | | | | |
| 10 | Nd-145 | 3.37% | 56.35% | 63 | Sb-125 | 0.10% | 97.66% | 116 | Sn-126 | 0.01% | 99.75% | | | | |
| 11 | Cs-135 | 2.74% | 59.09% | 64 | Tb-159 | 0.10% | 97.75% | 117 | Te-125 | 0.01% | 99.76% | | | | |
| 12 | Nd-143 | 2.64% | 61.73% | 65 | Sm-154 | 0.10% | 97.85% | 118 | Sn-120 | 0.01% | 99.76% | | | | |
| 13 | Xe-131 | 2.38% | 64.11% | 66 | Sr-90 | 0.10% | 97.95% | 119 | Sm-153 | 0.00% | 99.77% | | | | |
| 14 | Ru-102 | 2.21% | 66.31% | 67 | I-131 | 0.09% | 98.04% | 120 | Mo-96 | 0.00% | 99.77% | | | | |
| 15 | Sm-151 | 2.19% | 68.51% | 68 | Y-89 | 0.09% | 98.12% | 121 | Cs-136 | 0.00% | 99.77% | | | | |
| 16 | Mo-95 | 2.15% | 70.66% | 69 | Ba-138 | 0.08% | 98.20% | 122 | Sn-122 | 0.00% | 99.78% | | | | |
| 17 | Mo-98 | 1.89% | 72.55% | 70 | Pr-143 | 0.08% | 98.28% | 123 | Sn-124 | 0.00% | 99.78% | | | | |
| 18 | Ag-109 | 1.80% | 74.35% | 71 | Br-81 | 0.08% | 98.36% | 124 | Pm-148 | 0.00% | 99.79% | | | | |
| 19 | Ru-104 | 1.69% | 76.04% | 72 | Te-130 | 0.08% | 98.44% | 125 | As-75 | 0.00% | 99.79% | | | | |
| 20 | Mo-100 | 1.58% | 77.63% | 73 | In-115 | 0.08% | 98.51% | 126 | Xe-128 | 0.00% | 99.79% | | | | |
| 21 | Eu-153 | 1.56% | 79.19% | 74 | Te-128 | 0.07% | 98.59% | 127 | Ba-134 | 0.00% | 99.79% | | | | |
| 22 | Zr-93 | 1.27% | 80.45% | 75 | Cd-112 | 0.07% | 98.65% | 128 | Xe-135 | 0.00% | 99.80% | | | | |
| 23 | Ru-103 | 1.19% | 81.65% | 76 | Te-129m | 0.07% | 98.72% | 129 | Zr-90 | 0.00% | 99.80% | | | | |
| 24 | Pr-141 | 1.03% | 82.67% | 77 | Rb-87 | 0.06% | 98.78% | 130 | Nd-142 | 0.00% | 99.80% | | | | |
| 25 | I-129 | 0.97% | 83.65% | 78 | Kr-84 | 0.06% | 98.84% | 131 | Sn-115 | 0.00% | 99.80% | | | | |
| 26 | Zr-95 | 0.88% | 84.53% | 79 | Xe-133 | 0.05% | 98.89% | 132 | Te-126 | 0.00% | 99.80% | | | | |
| 27 | Zr-96 | 0.75% | 85.28% | 80 | Sb-121 | 0.05% | 98.94% | 133 | Sr-86 | 0.00% | 99.80% | | | | |
| 28 | Nd-146 | 0.70% | 85.98% | 81 | Te-127m | 0.05% | 98.99% | 134 | Gd-154 | 0.00% | 99.80% | | | | |
| 29 | Xe-132 | 0.69% | 86.67% | 82 | Pm-148m | 0.05% | 99.03% | 135 | Eu-152 | 0.00% | 99.80% | | | | |
| 30 | Pd-108 | 0.68% | 87.35% | 83 | Se-79 | 0.05% | 99.08% | 136 | Sb-124 | 0.00% | 99.80% | | | | |
| 31 | Nb-95 | 0.67% | 88.02% | 84 | Rh-105 | 0.05% | 99.13% | 137 | Te-122 | 0.00% | 99.80% | | | | |
| 32 | Ce-141 | 0.62% | 88.63% | 85 | Sm-150 | 0.04% | 99.17% | 138 | Kr-82 | 0.00% | 99.80% | | | | |
| 33 | Zr-91 | 0.61% | 89.24% | 86 | Sb-123 | 0.04% | 99.21% | 139 | Ge-76 | 0.00% | 99.80% | | | | |
| 34 | Zr-92 | 0.48% | 89.72% | 87 | Gd-155 | 0.03% | 99.24% | 140 | Ge-73 | 0.00% | 99.80% | | | | |
| 35 | Xe-134 | 0.48% | 90.20% | 88 | Sn-117 | 0.03% | 99.28% | 141 | Sn-116 | 0.00% | 99.80% | | | | |
| 36 | Ru-106 | 0.48% | 90.68% | 89 | Pm-149 | 0.03% | 99.31% | 142 | Te-124 | 0.00% | 99.80% | | | | |
| 37 | Sm-152 | 0.48% | 91.16% | 90 | Xe-136 | 0.03% | 99.34% | 143 | Ge-74 | 0.00% | 99.80% | | | | |
| 38 | Nd-148 | 0.46% | 91.62% | 91 | Pd-104 | 0.03% | 99.37% | 144 | La-138 | 0.00% | 99.80% | | | | |
| 39 | Cd-111 | 0.44% | 92.06% | 92 | Gd-158 | 0.03% | 99.40% | 145 | Ge-72 | 0.00% | 99.80% | | | | |
| 40 | Rb-85 | 0.43% | 92.49% | 93 | Ru-100 | 0.03% | 99.43% | 146 | In-113 | 0.00% | 99.80% | | | | |
| 41 | I-127 | 0.42% | 92.91% | 94 | Kr-85 | 0.03% | 99.46% | 147 | Gd-152 | 0.00% | 99.80% | | | | |
| 42 | La-139 | 0.42% | 93.33% | 95 | Sr-89 | 0.03% | 99.49% | 148 | Ba-135 | 0.00% | 99.80% | | | | |
| 43 | Pd-106 | 0.41% | 93.74% | 96 | Cd-114 | 0.02% | 99.51% | 149 | Xe-129 | 0.00% | 99.80% | | | | |
| 44 | Eu-155 | 0.35% | 94.08% | 97 | Sr-88 | 0.02% | 99.54% | 150 | Se-76 | 0.00% | 99.80% | | | | |
| 45 | Zr-94 | 0.32% | 94.40% | 98 | Sn-119 | 0.02% | 99.56% | 151 | Ru-99 | 0.00% | 99.81% | | | | |
| 46 | Sm-147 | 0.31% | 94.71% | 99 | Sm-148 | 0.02% | 99.58% | 152 | Nb-94 | 0.00% | 99.81% | | | | |
| 47 | Ce-142 | 0.29% | 95.00% | 100 | Se-82 | 0.02% | 99.59% | 153 | Cd-108 | 0.00% | 99.81% | | | | |
| 48 | Nd-150 | 0.28% | 95.28% | 101 | Ba-136 | 0.02% | 99.61% | 154 | Te-123 | 0.00% | 99.81% | | | | |
| 49 | Nd-147 | 0.26% | 95.54% | 102 | Ag-110m | 0.02% | 99.62% | 155 | Sr-87 | 0.00% | 99.81% | | | | |
| 50 | Cs-137 | 0.25% | 95.80% | 103 | Se-77 | 0.01% | 99.64% | 156 | Kr-80 | 0.00% | 99.81% | | | | |
| 51 | Y-91 | 0.20% | 96.00% | 104 | Kr-86 | 0.01% | 99.65% | 157 | Br-79 | 0.00% | 99.81% | | | | |
| 52 | Nd-144 | 0.19% | 96.19% | 105 | Eu-156 | 0.01% | 99.66% | 158 | Sn-114 | 0.00% | 99.81% | | | | |
| 53 | Kr-83 | 0.19% | 96.37% | 106 | Se-80 | 0.01% | 99.67% | 159 | Li-6 | 0.00% | 99.81% | | | | |

R : Absorption ratio of each nuclide
A : Accumulated absorption ratio

About 99.8% of the total FP neutron absorption comes from 165 major nuclides as shown in Table 1. The cross section data for these neutrons are stored in the JENDL-3.2 library. The contributions of other FP nuclides were found to be negligible so the calculation used only these 165 FP nuclides.

2.2 Computing Group Constants

The lumped group constants for the FP nuclides were generated as follows. The 100 group infinite dilution cross section of each individual FP nuclide was computed with the NJOY-94 [6] code. The energy group structure is the same as the JSDJ2 set and the scattering anisotropy is considered up to P3 components of Legendre expansion.

Atomic number densities of the FP nuclides generated from fission nuclides of $^{235}$U, $^{238}$U, $^{239}$Pu and $^{241}$Pu were independently computed by ORIGEN2 as a function of fuel burn-up. The lumped FP constants were then obtained as shown in equation (1) by averaging the infinite dilution cross sections with the atomic number densities based on the assumption that one fission produces one lumped FP.

$$\sigma^{X}_{Lamped\ FP}(B) = \frac{\sum_{i} N_{FP\,i}{}^{X}(B)\cdot\sigma_{i}}{\sum_{i} N_{FP\,i}{}^{X}(B)} \qquad (1)$$

Where $N^{X}_{FPi}(B)$ is the amount of FP nuclide i generated from fissile nuclide x (for $^{235}$U, $^{238}$U, $^{239}$Pu and $^{241}$Pu) at the burn-up B, and $\sigma_i$ is the cross section of FP nuclide i. The burn-up of the JOYO Mk-II driver fuel was assumed to be 10, 40, 63 and 90GWd/t. These FP group constants were added to the JSDJ2/JFTJ2 [4] set where the neutron self-shielding factor was set to be unity since the JOYO burn-up value did not affect the neutron self-shielding.

2.3 Verification of FP Constants

To verify the calculated lumped FP constants, the absorption microscopic cross section data was compared with the JFS-3-J3.2 [7] group constants used for the fast reactor core calculation. The lumped group constants for FP nuclides produced from $^{239}$Pu fission reaction at the 40GWd/t burn-up level are shown in Fig. 2. Both cross section curves agreed well. There exists little difference in The FP group constants did not vary significantly at four burn-up levels as shown in Fig. 3.

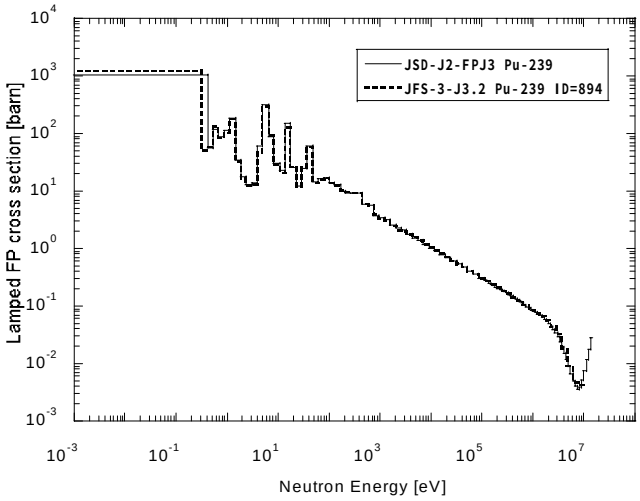

Fig.2 Lumped FP Cross-Section from $^{239}$Pu

The macroscopic absorption cross-section of the fuel materials was calculated using the RADHEAT-V3 [8] system. Calculations were done with and without FP. The comparison of the results are shown in Fig. 4. Some difference was observed in the lower energy region. However, both results agreed well above 1keV, which is important for analyzing a small fast reactor like JOYO because the absorption cross-section differed by only about 1 to 3% less than 0.3MeV.

## 3. The Effect of FP on Shielding Analysis

### 3.1 Shielding Calculation Method

In order to evaluate the effect of FP nuclides on the JOYO shielding analysis, the neutron flux distribution was calculated using the DORT [9] two-dimensional transport code. Table 2 lists the key elements of the calculation method. The neutron flux distributions were calculated in the RZ and XY-R θ geometry with the fixed source mode and neutron source distribution obtained from the JOYO Mk-II core management code system MAGI [10].

In the XY-R θ calculation, neutron leakage in the axial direction was considered as a pseudo-absorption to calculate the neutron flux distribution in the core center plane. This axial leakage term was obtained using the gradient of the neutron flux by the RZ calculation at 2.5 cm above and below the horizontal surface of the core.

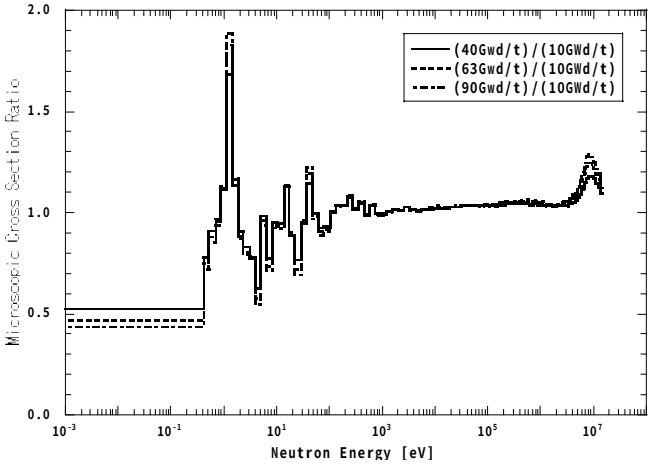


Fig.3 Burn-up Dependency of $^{239}$Pu Lumped FP

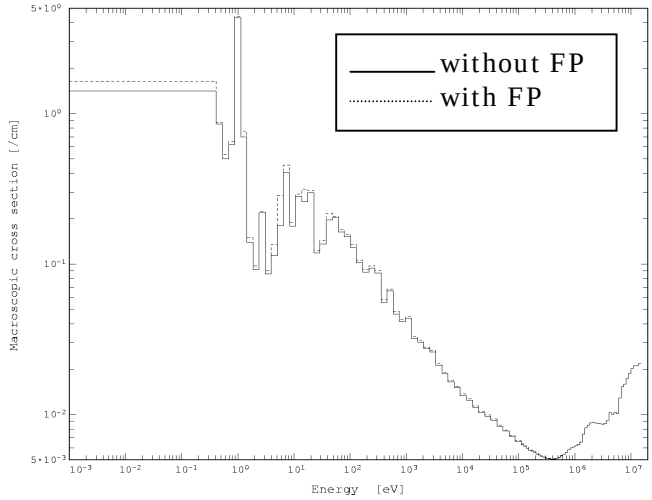


Fig.4 Macroscopic Absorption Cross-Section of Fuel Material

Table 2 Neutron Flux Calculation Conditions

| | |
|---|---|
| Transport code | DORT |
| Cross section calculation code | RADHEAT-V3 |
| Neutron cross section set | JSDJ2 |
| Self shielding factor set | JFTJ2 |
| $\gamma$ -ray production Library | Modified New-POPOP4 [11]<br>(including delayed fission $\gamma$ -ray) |
| No. of angular quadratures | S30 |
| Order of scattering anisotropy | P3 |

### 3.2 Effect on Neutron Flux

The comparison of the neutron spectra calculated with and without FP in the core center is shown in Fig. 5. The neutron flux above 1MeV decreases by about 2% when considering neutron absorption by FP nuclides. The ratio of total and fast (E>0.1MeV) neutron flux calculated along the core center plane in the RZ geometry with and without FP is shown in Fig. 6. The total neutron flux decreased by about 1.5% in in-vessel storage rack region when considering FP, where moderated neutrons would be captured by FP nuclides.

Note that at the position furthest away from the in-vessel spent fuel storage rack, the total and fast neutron flux decreased by only 2% when considering FP.

## 4. Conclusion

The fast reactor shielding calculations were improved by including the group constants of FP nuclides that were computed based on the JENDL-3.2 library. The effect of FP nuclides on neutron flux calculations was evaluated in JOYO, which has a hard neutron spectrum. It was found that the conventional calculations that ignored FP overestimated the neutron flux by about 2%.

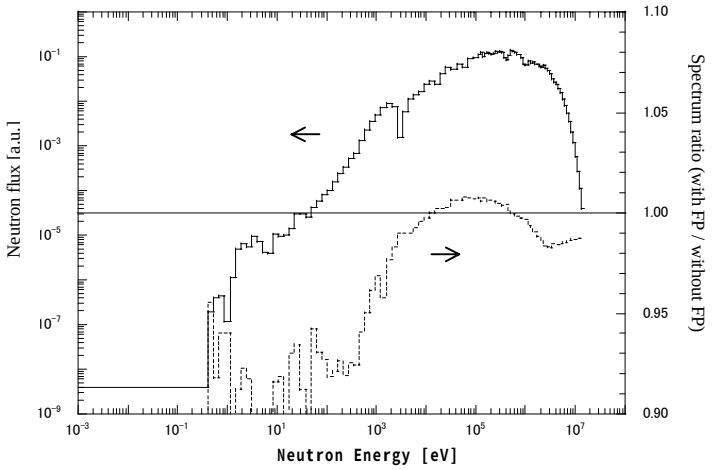


Fig.5 Comparison of Neutron Spectrum in the JOYO Core Center

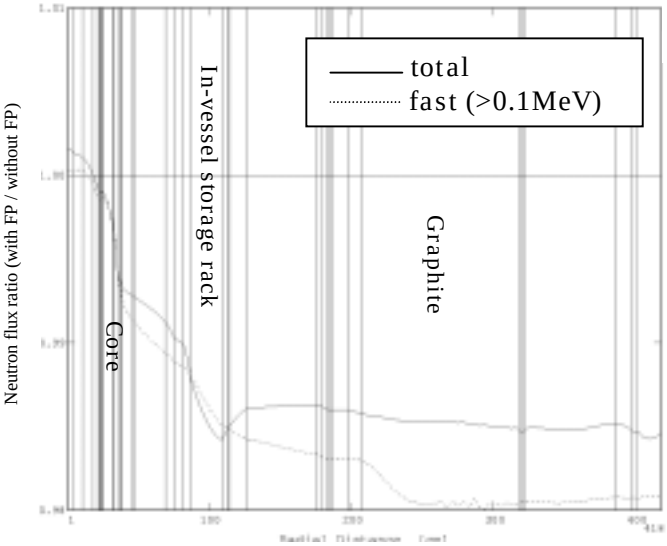


Fig.6 The Ratio of Neutron Flux with and without FP by RZ Calculation


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