Cross Sections for Actinide Burner Reactors*

Felix C. Difilippo
Oak Ridge National Laboratory
Oak Ridge, TN 37831-6364

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Send correspondence to:

Felix C. Difilippo
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6363

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CROSS SECTIONS FOR ACTINIDE BURNER REACTORS

Felix C. Difilippo
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6363, USA

Abstract: Recent studies have shown the feasibility of burning higher actinides (i.e., transuranium (TRU) elements excluding plutonium) in ad hoc designed reactors (Actinide Burner Reactors: ABR) which, because of their hard neutron spectra, enhance the fission of TRU. The transmutation of long-lived radionuclides into stable or short-lived isotopes reduces considerably the burden of handling high-level waste from either LWR or Fast Breeder Reactors (FBR) fuels. Because of the large concentrations of higher actinides in these novel reactor designs the Doppler effect due to TRU materials is the most important temperature coefficient from the point of view of reactor safety. Here we report calculations of energy group-averaged capture and fission cross sections as function of temperature and dilution for higher actinides in the resolved and unresolved resonance regions. The calculations were done with the codes SAMMY in the resolved region and URR in the unresolved regions and compared with an independent calculation.

Introduction

Neutronic and thermalhydraulic calculations have shown that it is possible to burn higher actinides in especially designed fast reactors which because of their particular hard neutron spectra enhance the fission of transuranic element (TRU). The analysis of a proposed Actinide Burner Reactor (ABR) plant (400 MWe) shows that it would be possible to burn in one year the amount of reprocessed TRU from eleven 1000 MWe Light Water Reactors (LWR) that had been operated for one year. The transmutation of long-lived radionuclides into stable or short-lived isotopes reduces considerably the burden of handling high-level waste from either LWR or Fast Breeder Reactors (FBR) fuels.

Detailed calculations of the capture and fission rates are essential to the aforementioned analysis whose accuracy depends on the careful preparation of the cross sections sets. Because of the large concentrations of higher actinides in these novel reactor designs, the Doppler effect due to TRU materials is the most important temperature coefficient from the point of view of reactor safety. Here we report calculations of energy group-averaged capture and fission cross sections as function of temperature and dilution for actinides in the resolved and unresolved resonance regions. The calculations were done with the codes SAMMY² in the resolved region, URR³ in the unresolved regions, and the ENDF/B-V evaluation of resonance parameters (single-level Breit Wigner formalism).

In order to define dilutions of higher actinides in realistic systems we had chosen, as an example, one of the cases analyzed in Ref. 1. Details of the unit fuel cell are summarized in Table 1 with information on the TRU isotopic composition.

Table 1. Parameters of a Typical ABR Fuel Cell

Fuel:	Np-Pu-Zr alloy (Inner Core, IC) Am/Cm-Pu-Y alloy (Outer Core, OC) (Volumetric Fraction $\alpha_F = 0.4031$) Density: 9.788 g/cm ³ (IC), 11.700 g/cm ³ (OC)
Isotopes ^a :	IC: ²³⁷ Np, 58%; Pu, 22%; Zr, 20% OC: Am/Cm, 60%; Pu, 35%; Y, 5%
Coolant:	Na, $\alpha_c = 0.3137$
Fuel Pin Radius:	2mm

^aWeight percent.

The dilution of the absorbing atom is parameterized in different ways according to the approximations used to compute the average cross section

$$\bar{\sigma}_{p,i}(T) = \frac{\int_{E_{i+1}}^{E_i} \phi(E) \sigma_p(E, T) dE}{\int_{E_{i+1}}^{E_i} \phi(E) dE} \quad (1)$$

for the process p of the resonant absorber. In the Bondarenko treatment ϕ is approximated as proportional to $1/\sum_i(E)$ where the total cross section $\sum_i = n(\sigma_i + \sigma_s)$, n is the density of the resonant absorber with total cross section σ_i , and σ_s is the scattering cross section of the rest of the atoms of the mixture per absorbing atom. In other, more accurate, approximations (like the treatment of the resolved region in the AMPX system) $\phi(E)$ is computed in terms of the scattering cross sections of specific atoms per absorbing atom and their respective masses. Table 2 shows values of σ_s for the cell of Table 1.

Table 2. Scattering Cross Section Per Absorbing Atom for the Cells of Table 1

Absorbing Atom	Homogeneous Cell σ_s (barn)		Heterogeneous Cell ^(c) σ_s (barn)	
	All Isotopes	Sodium	Isolated Lump	Interactive Lumps
²³⁷ Np	30.2	4.0	173.0	52.0
²⁴¹ Am ^(a)	48.1	5.4	237.0	71.0
²⁴³ Am	118.0	11.8	520.0	156.0
²⁴³ Cm ^(b)	52500.0	4836.0	210,000.0	63300.0
²⁴⁴ Cm	295.0	28.2	1230.0	372.0
²⁴⁸ Cm	5567.0	515.0	22800.0	6800.0

^a87.63% Am in Am/Cm alloy; 68.7% ²⁴¹Am; 31.3% ²⁴³Am, (from PWR).

^b0.55% ²⁴³Cm; 94.28% ²⁴⁴Cm; 5.17% ²⁴⁸Cm, (from PWR).

^cMean chord: 0.4cm; Dancoff Coefficient: 0.7. σ_s denotes the escape cross section in the Wigner's rational approximation.

Calculations of Cross Sections and Bondarenko Factors

The codes SAMMY² (for the resolved region) and URR³ (for the unresolved region) were used to compute energy group-averaged cross sections as functions of dilution and temperature. The Bondarenko approximation for the energy dependence of the flux was used in both regions, so a universal parameter σ_s (potential scattering cross section of the rest of the mixture of the isotopes per absorbing atom) appears

without any reference to the masses of the isotopes or the moderation process within the width of the resonances.

With the code SAMMY, we computed the point cross sections as functions of temperature with parameters from the ENDF/B-V evaluation which provides single-level Breit-Wigner resonance parameters and assumes $g=0.5$ for all resonances. Because SAMMY is based on an R -matrix theory description of the cross sections that require the spin of each resonance, we arbitrarily divided the set of resonances into a large number of spin families in order to eliminate the interference between resonances and to be consistent with the single-level evaluation.

The next step was to average the temperature-dependent cross section with the Bondarenko prescription:

$$\bar{\sigma}_{p,i}(\sigma_o, T) = \frac{\int_{E_{i-1}}^{E_i} \sigma_p(E, T) w(E, T) dE}{\int_{E_{i-1}}^{E_i} w(E, T) dE} \quad (2)$$

where the weighting function w is

$$w(E, T) = \frac{1}{\sigma_t(E, T) + \sigma_o} \quad (3)$$

The Bondarenko factor for process p , group i , temperature T and dilution σ_o was also computed according to its definition

$$B_{p,i}(T, \sigma_o) = \frac{\bar{\sigma}_{p,i}(\sigma_o, T)}{\bar{\sigma}_{p,i}(\infty, T)} \quad (4)$$

We calculated the cross sections in the unresolved region with the code URR³ which computes the probability distribution of the cross section at an input energy while playing Monte Carlo in the surrounding energy region with the average resonance parameters (also from the ENDF/B-V evaluation). After the calculation of the probability table, the code evaluates the shielded averaged cross sections and the Bondarenko factors. Using the ergodic hypothesis, these parameters are considered average values for the energy region around the input energy; this way of calculating the parameters is faster than the calculation of a ladder of resonances in a broad energy group. The use of ENDF/B-V data allowed us to compare the numerical approach presented here with similar calculations performed with the system of codes AMPX (used by the nuclear industry) which in its present version do not include a R -matrix description of the cross sections. To compare both approaches we select the unresolved region, because it is more important than the resolved region for the design described in Table 1, and the actinides ²³⁷Np, ²⁴¹Am, ²⁴⁴Cm, because they are the most shielded of each actinide element.

Figure 1 compares the AMPX results with the URR calculations for infinite diluted capture and fission cross sections of ²³⁷Np, ²⁴¹Am, ²⁴⁴Cm, in their respective unresolved region: 130eV-40keV; 50eV-10keV and 525eV-10keV.

The comparison is very good; ²³⁷Np exhibits the structure of the subthreshold fission so a suitable energy grid would have been necessary to fully describe it which was not intended in the calculations of Fig. 1.

Fig. 1. Infinite diluted actinide cross sections in the unresolved region. Histogram calculated with the AMPX system, points calculated with the URR code.

The Bondarenko factors for the capture process at $T=293^\circ\text{K}$ and $\sigma_o=100$ barns are shown in Fig. 2 computed with the AMPX system and the URR code. Again the comparison shows consistency.

Fig. 2. Bondarenko factors for the capture process. Histogram calculated with the AMPX system; points calculated with the URR code.

Conclusions

Because of the major presence of higher actinides in ABRs of innovative design and their consequent impact on the Doppler coefficient of reactivity, effective capture and fission cross sections and Bondarenko factors were calculated for ^{237}Np , ^{241}Am , ^{243}Am , ^{243}Cm , ^{244}Cm , and ^{245}Cm in the resolved and unresolved resonance regions. The calculations were done with the computer codes SAMMY and URR, which are codes mainly used for the analysis of experimental data. This new approach has the following advantages: (1) It uses the same numerical tool already used in the analysis of experimental data, i.e., the method is more transparent to the original data; (2) improvements in the analysis of experimental data (e.g., the R matrix analysis performed by SAMMY) can be introduced directly in the preparation of the cross sections; and (3) the ergodic hypothesis implemented in the code URR is faster than explicit calculations of ladders of resonances in a broad energy group. Extensive calculations as functions of dilution factors and temperature were made with resonance parameters from ENDF/B-V (single Breit-Wigner resonance parameters) that allowed direct comparison of the results with an independent calculation with the AMPX system.

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