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EVALUATION AND ADJUSTMENT OF ACTINIDE CROSS SECTIONS USING INTEGRAL DATA MEASURED AT FCA

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Abstract : The actinide integral experiments were conducted in the IX series assemblies (standard neutron spectrum assemblies) of FCA. The 20 group fission and capture cross sections of actinide nuclides processed from JENDL-2 library were evaluated and adjusted using these integral data. These adjusted cross sections were checked using the integral data measured in the assemblies other than the IX series. It is concluded that the adjusted cross sections can be generally used in fast spectrum.

(actinide cross sections, cross section evaluation and adjustment, integral experiment, least squares fitting, fast reactor, fission rate ratio, sample reactivity worth, neutron spectrum, higher actinide, fast critical facility)

Introduction

To test and improve the reliability of the fission and capture cross sections of higher actinides, the integral experiment program was conducted using the fast critical facility FCA.[1,2]

The program can be devided into four parts; (1) experiments for the characteristics of neutron fields where integral measurement were carried out, (2) fission rate and sample worth measurements of higher actinides, (3) evaluation of cross section data using integral data and (4) adjustment of cross section data by the least squares fitting method.

In this paper, 20 group fission and capture cross sections of actinide nuclides processed from JENDL-2 library were evaluated and adjusted using integral data. It was also shown that the adjusted data could be generally used in fast spectrum.

Integral Measurement

Neutron Field Characteristics

Seven assemblies were built so as to change spectrum shift systematically.[3]

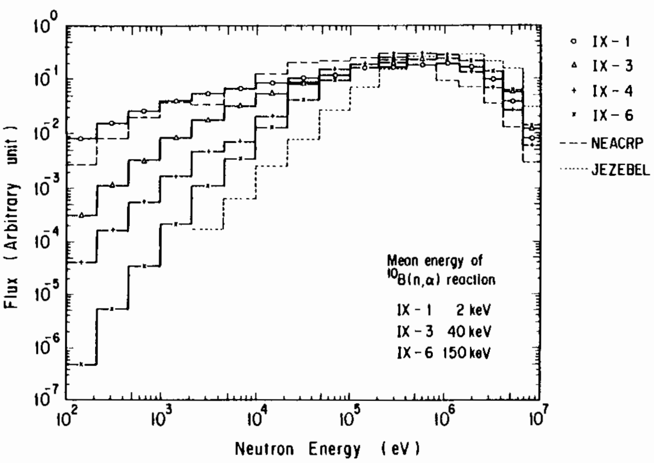

Fig.1 Calculated neutron spectra at the center of FCA IX assemblies

The assemblies IX-1 to IX-6 were composed with 93% enriched metal uranium and diluent material of graphite or stainless steel for adjusting neutron spectrum in lower energy region or in higher energy region, respectively. The assembly IX-7 was composed with 20% enriched metal uranium. The core cell patterns were designed to be symmetric within one-drawer unit cell so as to make calculation simple.

The mean energies of B-10 (n,α) reaction are about 2keV, 40keV and 150keV for the assembly IX-1, IX-3 and IX-6, respectively. Higher component of neutron spectrum depends on the diluent material. Typical center spectra of these assemblies are shown in Fig.1 for comparison with those of NEACRP and JEZEBEL.

At each assembly, k_{eff}, fission rates and sample reactivity worths of conventional materials were measured to check the reliability of neutron fields calculation.

Actinide Integral Measurement

Fission rate ratios and small sample reactivity worths of higher actinides were measured.

Fission rates of Np-237, Pu-238, Pu-239, Pu-242, Am-241, Am-243 and Cm-244 relative to fission in U-235 were measured at the core center using parallel-plate type fission chamber of which the deposit mass was determined within ± 1.5% error (± 3.0%, for Cm-244)[4]. After the corrections for fission in impurity isotopes and for flux perturbation caused by the introduction of a fission chamber, the uncertainties of fission rate measurements were about 2% (4% for Cm-244). The fission rate ratios measured at the core center are shown in Fig.2.

Sample reactivity worth measurements were carried out for the isolated actinide nuclides of Np-237, Pu-238, Pu-240, Am-241 and Am-243. Sample materials, about 15 to 20 grams in weight, were loosely packed oxide powders contained in double cylindrical stainless-steel capsules (inner diam.; 1cm, length; ∿ 8cm). After making corrections for the reactivity contributions from a capsule, oxygen and impurity isotopes, the reactivity worths of major actinides were obtained. The reactivity worth ratios measured at the core center are shown in Fig.3.

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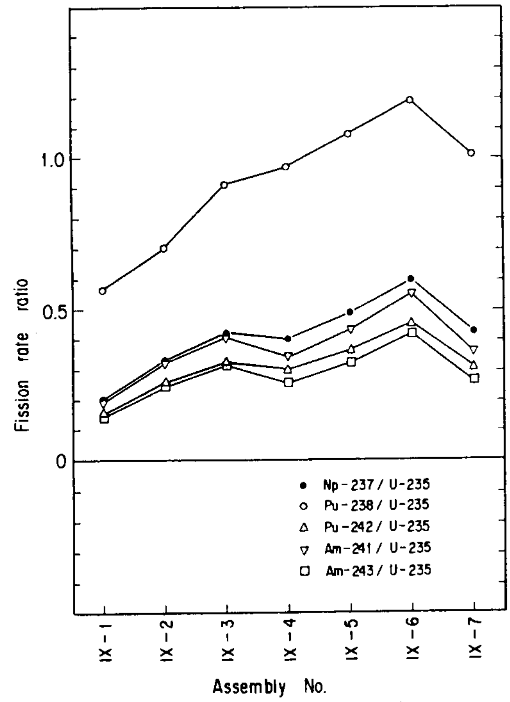

Fig.2 Actinide fission rate ratios measured in FCA IX assemblies

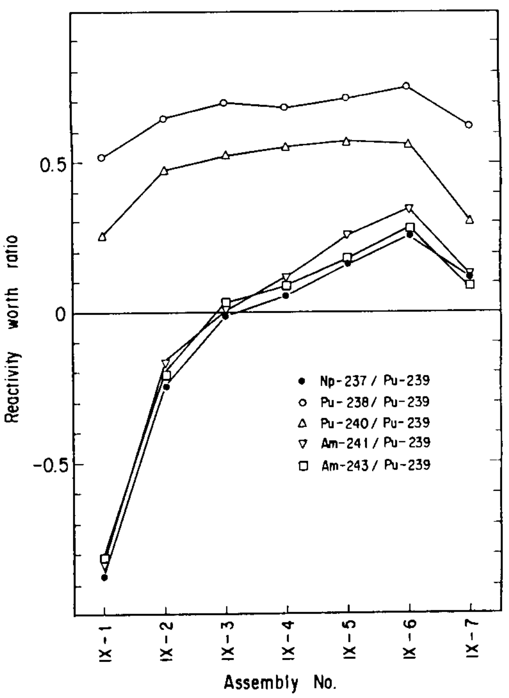

Fig.3 Actinide sample worth ratios measured in FCA IX assemblies

Evaluation and Adjustment Method

Neutron Field Calculation

Neutron fields were calculated with JENDL-2 library. The collision probability code, SP-2000, was used to calculate a fine group (1970 groups) fundamental-mode spectrum. The 20 group cell averaged, self shielded cross sections were generated using the fine group spectrum. Real and adjoint fluxes were obtained from the two dimensional transport calculation using the R-Z model of each assembly.

Evaluation of actinide cross sections

The higher actinides fission and capture cross sections of JENDL-2 library were collapsed into 20 groups. In this collapsion, a fine group fundamental-mode spectrum of the assembly IX-4 was used as a weighting function. Higher actinide cross sections were evaluated by comparing calculated and measured values. In the calculation of these integral values, the real and adjoint fluxes and the 20 group cross sections mentioned above were used.

Adjustment of actinide cross sections

The sensitivity coefficients for the integral data were calculated with the generalized perturbation theory. The group cross sections were adjusted by the least squares fitting method with the following formula.[5]

$$\tilde{\sigma}_i = \hat{\sigma}_i + \Delta\hat{\sigma}_i \sum_{j}^{n} \sum_{k}^{m} \sum_{l}^{m} \rho_{ij} A_{kj} (B^{-1})_{kl} M_l \qquad (1)$$

where

- n : number of group cross sections to be adjusted
- m : number of integral data used for adjusting cross sections
- $\hat{\sigma}_i$: cross section in group i
- $\Delta\hat{\sigma}_i$: standard deviation of $\hat{\sigma}_i$
- ρ_{ij} : (i, j) element of the correlation matrix $\boldsymbol{\rho}$ of cross section
- A_{kj} : (k, j) elememt of the sensitivity coefficient matrix **A** of integral data
- $(B^{-1})_{kl}$: (k, l) element of the inverse matrix of **B** ($\mathbf{B} = \mathbf{E} + \mathbf{A}\boldsymbol{\rho}\mathbf{A}^T$)
- M_l : l element of the vector **M** of the difference between the measured and calculated values of integral data

The calculation flow for the actinide integral experiment and cross section adjustment is shown in Fig.4.

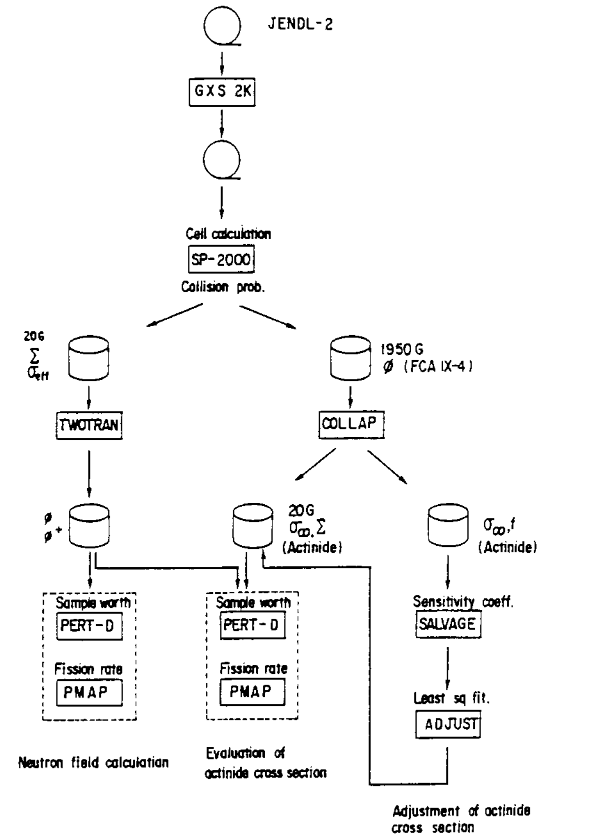

Fig.4 Calculation flow for actinide integral experiment and cross section adjustment

Results

Neutron field calculation

To test the reliability of the neutron field calculation, calculated (C) and experimental (E) values of reactor physics parameters are compared and the results are shown in Fig.5 as C/E values. The agreement between calculated and experimental values are satisfactory and we can conclude the high reliability of neutron field calculation.

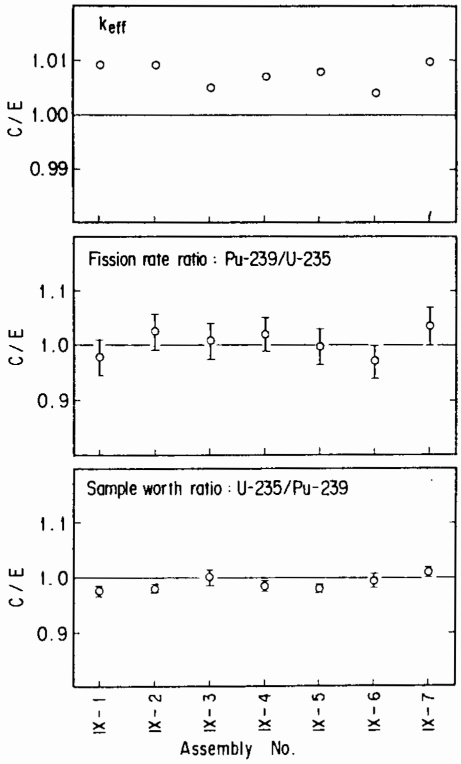

Fig.5 Comparison of values between calculated (C) and measured (E) of reactor physics parameters

Evaluation for actinide cross sections

Figure 6 shows the comparison of values between JENDL-2 calculations (C) and experiments (E) as C/E values.

(1) Fission rate ratios

The calculated and experimental values of Np-237 and Pu-238 agree within experimental errors (± 2%). For those of other nuclides, the calculation gives 6 to 15% larger values than the experiments.

(2) Sample reactivity worth ratios

The harder a neutron spectrum becomes, the larger the descrepancy of C/E from 1 is except in the assembly IX-7. The descrepancy between calculated and experimental values becomes largest in the assembly IX-7 except for Pu-240. In the case of Pu-240, the calculation gives small worths and C/E values are 0.4 to 0.8 in the assemblies IX-1 to IX-6.

Adjustment for actinide cross sections

The fission cross section of Am-243 was corrected 15% in the energy range above 100keV. Correction for the fission cross sections of the other nuclides were less than 15%. On the other hand, the correction factors for the capture cross sections of some nuclides were about two in some energy range.

The comparison between the measured data (E) and the calculated values (C) using the adjusted cross sections is shown in Fig.7. The adjustment effect is clearly shown by comparing Fig.6 and Fig.7.

For the fission rate ratios, the calculation with the adjusted actinide cross sections agrees with the experimental values

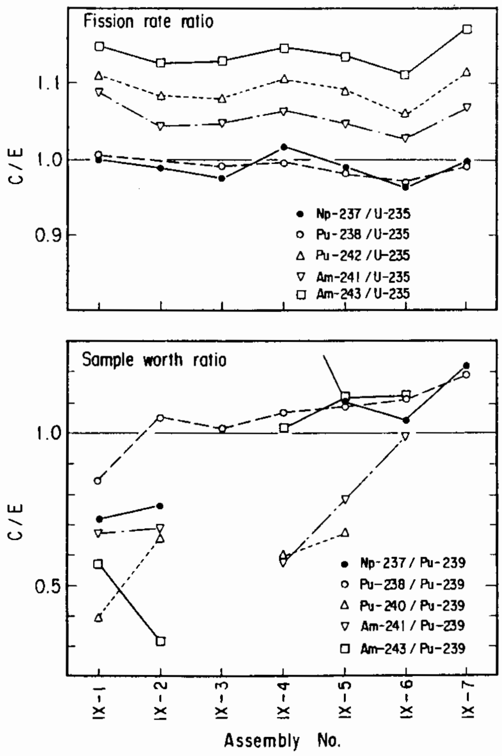

Fig.6 Comparison of values between calculated (C) and measured (E) of actinide integral data (JENDL-2 was used for the calculation)

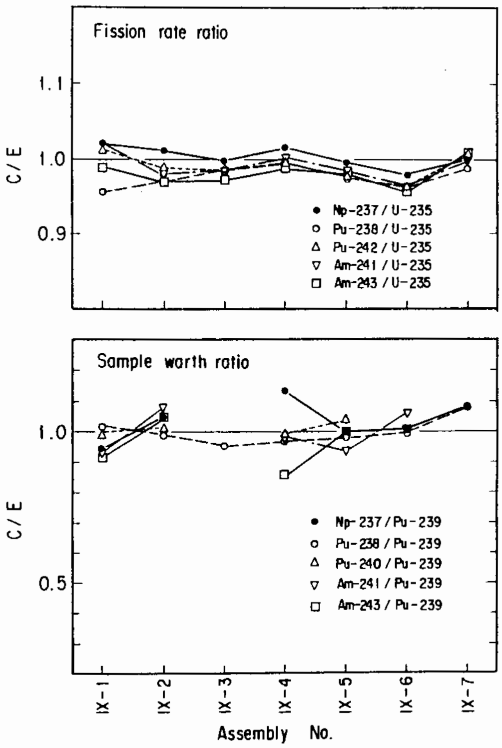

Fig.7 Comparison of values between calculated (C) and measured (E) of actinide integral data (Adjusted JENDL-2 was used for the calculation)

within experimental errors except for in the assembly IX-6. The C/E values of the sample reactivity worth ratios are 0.9 to 1.1.

We can conclude, therefore, that the adjustment of higher actinides was successfully done.

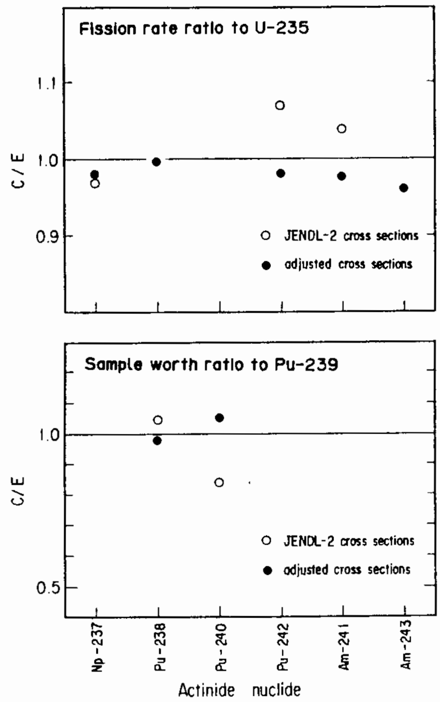

Fig.8 Comparison of C/E values between the calculated values (C) using JENDL-2 cross sections and adjusted cross section for actinide integral data (E) measured in the assembly X-1

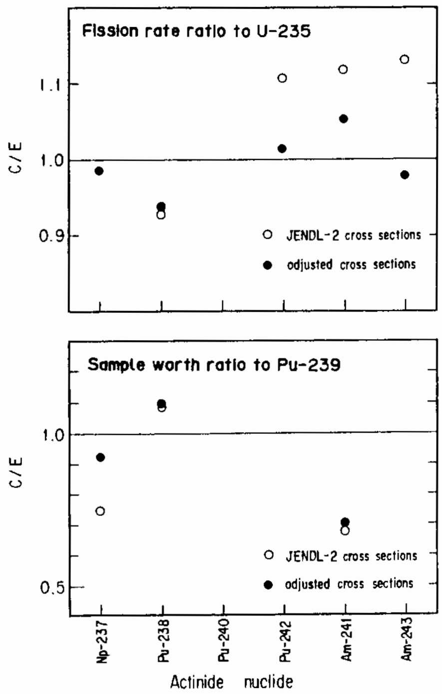

Fig.9 Comparison of C/E values between the calculated values (C) using JENDL-2 cross sections and adjusted cross sections for actinide integral data (E) measured in the assembly XI-1

Application of the adjusted actinide cross sections

The reliability of the adjusted cross sections was tested for the actinide integral data measured in FCA assemblies X-1 and XI-1. The assembly X-1 is thè mockup for "JOYO Mark-II" core and the assembly XI-1 is the mockup for the large scale commercial FBR core. Neutron spectrum of the assembly XI-1 is softer than those of FCA IX assemblies. The C/E values for the actinide integral data were compared with using JENDL-2 and using the adjusted cross sections in Fig.8 and Fig.9. The adjusted cross sections give a better agreement between calculated and experimental values than the original ones.

In the above mentioned cross section adjustment, these integral data measured in the assemblies X-1 and XI-1 were not used. Even though, the calculation using the adjusted cross sections gives a better agreement with the experimental values. Therefore, it is concluded that the adjusted data can be used generally in fast spectrum.

Conclusion

The 20 group cross sections of higher actinides processed from JENDL-2 library were evaluated and adjusted using integral measurements in FCA. The adjusted cross sections were tested for the fission rate ratios and sample reactivity worths measured in the assemblies X-1 and XI-1.

The calculation based on JENDL-2 library agrees with the measured fission rate ratios of Np-237 and Pu-238 and overestimates those of the other nuclides in the IX series assemblies. When the JENDL-2 library was used for the calculation of sample reactivity worth ratios, C/E values show a spectrum dependency.

The calculation using the adjusted actinide cross sections better agrees with the experiments in FCA IX assemblies than that based on the orignal ones. The adjusted data were tested in the other than IX series assemblies and gave better results than the original JENDL-2.

The adjusted data given in this paper are preliminary ones because, in the adjustment procedure, we didn't consider the effects caused by the experimental errors of U-235 fission rate and of Pu-239 reactivity worth and the heterogeneity effects associated with the measurement of the sample reactivity worth.

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