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JAPAN ATOMIC ENERGY RESEARCH INSTITUTE
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Editor's Note

This is a collection of reports which have been submitted to the Japanese Nuclear Data Committee at the committee's request. The request was addressed to the following individuals who might represent or be in touch with groups doing researches related to the nuclear data of interest to the development of the nuclear energy program.

Although the editor tried not to miss any appropriate addressees, there may have been some oversight. Meanwhile, contribution of a report rested with discretion of its author. The coverage of this document, therefore, may not be uniform over the related field or research.

In this progress report, each individual report is generally reproduced as it was received by the JNDC secretariat, and editor also let pass some simple obvious errors in the manuscripts if any.

This edition covers a period of July 1, 1985 to June 30, 1986. The information herein contained is of a nature of "Private Communication". <u>Data contained in this report should not be quoted without the author's permission.</u>

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ELEMENT S	A	QUANTITY	ENERGY MIN	MAX	LAB	TYPE	DOCUMENTATION REF	VOL	PAGE	DATE	COMMENTS
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CR	052	N, 2N	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. FIG GIVEN
CR	052	N, PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG. NDG
FE		TOTAL	30+5	22+6	TOH	EXPT-PROG	NEANDC-J122		97	SEP 86	ISHIKAWA+.DYNAMITRON,TRANS. NDG
FE		DIFF ELASTIC	14+7	18+7	TOH	EXPT-PROG	NEANDC-J122		93	SEP 86	YABUTA+.DYNAMITRON. NDG
FE		DIFF INELAST	14+7	18+7	TOH	EXPT-PROG	NEANDC-J122		93	SEP 86	YABUTA+.DYNAMITRON. NDG
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NI		DIFF ELASTIC	14+7	18+7	TOH	EXPT-PROG	NEANDC-J122		93	SEP 86	YABUTA+.DYNAMITRON. NDG
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NI	060	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3

ELEMENT S	A	QUANTITY	ENERGY MIN	MAX	LAB	TYPE	DOCUMENTATION REF	VOL	PAGE	DATE	COMMENTS
CU		TOTAL	30+5	22+6	TOH	EXPT-PROG	NEANDC-J122		97	SEP 86	ISHIKAWA+.DYNAMITRON,TRANS. NDG
CU	063	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
CU	063	N, 2N	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG. NDG
CU	065	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
ZR	090	N, 2N	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG TO ZR-89M. NDG
ZR	090	N, PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. TO Y-90M. FIG GIVEN
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ZR	091	N, PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. TO Y-91M. FIG GIVEN
ZR	091	N,N PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. TO Y-90M. FIG GIVEN
ZR	092	N, PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. NDG
ZR	092	N,N PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. TO Y-91M. NDG
ZR	094	N, PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG. NDG
ZR	094	N,N PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. NDG
ZR	094	N, ALPHA	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. NDG
ZR	096	N, 2N	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		20	SEP 86	IKEDA+.ACT SIG. NDG
MO	092	N, 2N	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG TO MO-91M. NDG
MO	092	N, ALPHA	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG TO ZR-89M. FIG GIVEN
MO	092	N,N ALPHA	13+7	15+7	NAG	EXPT-PROG	NEANDC-J122		81	SEP 86	KATOH+.C-W. ACT SIG IN FIG
MO	094	N, PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG TO NB-94M. NDG
MO	097	N, PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG TO NB-97M. NDG

ELEMENT S	A	QUANTITY	ENERGY MIN	MAX	LAB	TYPE	DOCUMENTATION REF	VOL	PAGE	DATE	COMMENTS
MO	098	N,N PROTON	13+7	15+7	JAE	EXPT-PROG	NEANDC-J122		23	SEP 86	KONNO+.ACT SIG TO NB-97M. NDG
MO	098	N, ALPHA	13+7	15+7	NAG	EXPT-PROG	NEANDC-J122		81	SEP 86	KATOH+.C-W. ACT SIG IN FIG
SN	118	DIFF ELASTIC	14+7	18+7	JAE	EXPT-PROG	NEANDC-J122		9	SEP 86	CHIBA+.TANDEM TOF,D-D N. NDG
SN	118	DIFF INELAST	14+7	18+7	JAE	EXPT-PROG	NEANDC-J122		9	SEP 86	CHIBA+.TANDEM TOF,D-D N.TO 2+,3-.NDG
SM	148	RESON PARAMS	40+1	90+3	JAE	EXPT-PROG	NEANDC-J122		8	SEP 86	MIZUMOTO+.LINAC,TRANS.70+ LVLS.NDG
SM	148	STRNTH FNCTN	40+1	90+3	JAE	EXPT-PROG	NEANDC-J122		8	SEP 86	MIZUMOTO+.LINAC,TRANS.S0=3.8+-0.7
GD	152	N, GAMMA	MAXW		JAE	EXPT-PROG	NEANDC-J122		37	SEP 86	TOMINAGA+.ACT SIG=690+-30B
GD	153	N, GAMMA	MAXW		JAE	EXPT-PROG	NEANDC-J122		37	SEP 86	TOMINAGA+.ACT SIG=32000+-4000B
HF	174	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
HF	176	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
HF	177	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
HF	178	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
HF	179	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
HF	180	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
TH	232	RESON PARAMS	22+1	69+1	KTO	EXPT-PROG	NEANDC-J122		62	SEP 86	KOBAYASHI+.LINAC.WN,WG GIVN FOR 4RES
U	235	NONELA GAMMA	10-5	20+7	NIG	EVAL-PROG	NEANDC-J122		87	SEP 86	IIJIMA+.STAT+PREEQ CALC.FOR JENDL-3
U	235	DELAYD NEUTS	PILE		JAE	EXPT-PROG	NEANDC-J122		34	SEP 86	KANEKO+.INTEGRAL EXP. BETA=0.00676
U	235	FISS PROD GS	FAST		JAE	EVAL-PROG	NEANDC-J122		12	SEP 86	.DECAY HEAT CFD EXP DATA IN FIG
U	235	FISS PROD BS	FAST		JAE	EVAL-PROG	NEANDC-J122		12	SEP 86	.DECAY HEAT CFD EXP DATA IN FIG
U	236	FISSION	70+5	70+6	TOH	EXPT-PROG	NEANDC-J122		99	SEP 86	KANDA+.RATIO U236NF/U235NF IN FIG

ELEMENT S A	QUANTITY	ENERGY MIN	ENERGY MAX	LAB	TYPE	DOCUMENTATION REF	VOL	PAGE	DATE	COMMENTS
U 238	FISS PROD GS	FAST		JAE	EVAL-PROG	NEANDC-J122		12	SEP 86	.DECAY HEAT CFD EXP DATA IN FIG
U 238	FISS PROD BS	FAST		JAE	EVAL-PROG	NEANDC-J122		12	SEP 86	.DECAY HEAT CFD EXP DATA IN FIG
NP 237	FISSION	70+5	70+6	TOH	EXPT-PROG	NEANDC-J122		99	SEP 86	KANDA+.RATIO NP237NF/U235NF IN FIG
PU 239	FISS PROD GS	FAST		JAE	EVAL-PROG	NEANDC-J122		12	SEP 86	.DECAY HEAT CFD EXP DATA IN FIG
AM 243	FISSION	70+5	70+6	TOH	EXPT-PROG	NEANDC-J122		99	SEP 86	KANDA+.RATIO AM243NF/U235NF IN FIG
CF 250	EVALUATION	10-5	20+7	JAE	EVAL-PROG	NEANDC-J122		11	SEP 86	NAKAGAWA.PUBLISHED AS JAERI-M 86-086
CF 251	EVALUATION	10-5	20+7	JAE	EVAL-PROG	NEANDC-J122		11	SEP 86	NAKAGAWA.PUBLISHED AS JAERI-M 86-086
MANY	N, 2N	14+7	15+7	JPN	THEO-PROG	NEANDC-J122		43	SEP 86	HORIBE+.EMPIRICAL FORM CFD EXP.TABLE
MANY	N, PROTON	14+7	15+7	JPN	THEO-PROG	NEANDC-J122		43	SEP 86	HORIBE+.EMPIRICAL FORM CFD EXP.TABLE
MANY	N, ALPHA	14+7	15+7	JPN	THEO-PROG	NEANDC-J122		43	SEP 86	HORIBE+.EMPIRICAL FORM CFD EXP.TABLE

The content table in the CINDA format was compiled by the JNDC CINDA group;

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I. Electrotechnical Laboratory

Quantum Technical Division

I-1 Measurement of $^{27}Al(n,p)^{27}Mg$ Activation Cross Section at Neutron Energies of 15.2, 15.9 and 17.0MeV

K.Kudo, T.Kinoshita, H.Hino and Y.Kawada

The reaction $^{27}Al(n,p)^{27}Mg$ provides advantageous features as one of activation reactions for reactor dosimetry, in spite of the relatively short half life 9.462 min. for its precise activity measurement.

For last three years, our group concentrated on its energies on the cross section measurement of the $^{27}Al(n,\alpha)^{24}Na$ reaction, which is one of the standard data recommended by IAEA, as reported in the last progress report(1). For the determination of the $^{27}Al(n,p)^{27}Mg$ cross section, it seems experimentally advantageous and easy to perform the relative activity measurement of ^{27}Mg to the ^{24}Na activity by referring to the $^{27}Al(n,\alpha)^{24}Na$ cross section.

The aluminum disc samples(25.4mm dia.x0.3mm^{t}) were irradiated for five hours with monoenergetic neutrons of known energy between 15 and 17MeV produced by the $T(d,n)^{4}He$ reaction. The time variation of the neutron intensity was carefully monitored for the relatively short half life of ^{27}Mg compared to that of ^{24}Na(15 hours).

The γ counting of the 844keV from ^{27}Mg (γ yield 73%)(2) was performed with a well-calibrated pure Ge detector(64.9cc) shielded by a lead box, and the γ rays of 1.37 and 2.75Mev from ^{24}Na also measured in the same run simultaneously.

The cross section of the $^{27}Al(n,p)^{27}Mg$ can be derived by using the following expression,

$$\sigma_p = \sigma_\alpha \cdot \frac{A_p\ K_\alpha\ C_\alpha\ (\ 1-EXP(-\lambda_\alpha\ T))}{A_\alpha\ K_p\ C_p\ (\ 1-EXP(-\lambda_p\ T))} \quad ,$$

where the suffix p and α show the $^{27}Al(n,p)$ and $^{27}Al(n,\alpha)$ reaction respectively,and σ,A,K,C,$\lambda$and T are the cross section,activity at the reference time, correction for neutron time variation,correction for neutron irradiation,decay constant and irradiation time respectively.

The present results are summarized in Table 1 and compared with those from ENDF/B-V. So far our results are 3 to 14% below the ENDF/B-V evaluation in this energy region, but rather consistent with the recent work by Ryves et al[(3)]. The final analysis is in progress in the energy range of 14 to 20MeV.

Table 1 Activation cross sections of $^{27}Al(n,p)^{27}Mg$ reaction

energy(MeV)	present results(mb)	ENDF/B-V(mb)
15.21	66.7±1.9	68.6
15.88	56.8±1.7	63.6
16.98	47.5±2.4	55.3

References:

(1)K.Kudo,T.Michikawa,T.Kinoshita,Y.Hino and Y.Kawada,NEANDC(J)-116/U, 5(1985).

(2)Table of Isotopes 7th Edition (1978).

(3)T.B.Ryves,P.Kolkowski and K.J.Zieba,J.Phys. G,**4**,1783(1978).

II. JAPAN ATOMIC ENERGY RESEARCH INSTITUTE

A. Linac Laboratory, Department of Physics

II-A-1

Measurements of gamma-ray production cross sections.

M. Mizumoto, Y. Yamanouchi, M. Sugimoto, S. Chiba,

Y. Furuta and Y. Kawarasaki

The new detector system was installed to measure gamma-ray production cross sections from the $(n,n'\gamma),(n,\gamma)$ reactions at the JAERI Tandem Accelerator. The gamma-ray detector consists of a 7.6 cm diameter x 15.2 cm long NaI(Tl) detector surrounded by a 25.4 cm diameter x 25.4 cm long NaI(Tl) annular detector. They are set in a heavy shield composed of lead and borated paraffin and show bar. This detector system can be rotated around the sample to measure gamma-ray angular distributions. The gamma-ray spectrum of the $^{27}Al(n,n'\gamma)\,^{27}Al$ was measured as a test experiment. Neutrons were produced from the $D(d,n)^3He$ reaction. To determine the response functions of this anti-Compton NaI detector, the gamma-ray spectra from the standard sources such as ^{60}Co, ^{137}Cs and ^{88}Y and the reaction gamma-rays of $^{12}C(n,n'\gamma)^{12}C$ and $^{16}O(n,n'\gamma)^{16}O$ were obtained.

II-A-2

Neutron Resonance Parameters of ^{148}Sm

M. Mizumoto and Zhao Wen Rong*

The transmission data of ^{148}Sm were obtained by the neutron time-of-flight facility at the 120 MeV JAERI Linac. There have been no previous resonance data available for this isotope. The sample used was the oxide powder of ^{148}Sm enriched to 96.49 %. The measurements were carried out with a ^{6}Li-glass detector using a 55 m flight path. The transmission data were analyzed with a multi-level Breit-Wigner formula in a least squares fitting program. Resonance energies and neutron widths for more than 70 resonances of ^{148}Sm were newly determined in this experiment from 40 eV to 9000 eV. The s-wave strength function was obtained to be $(3.8 \pm 0.7) \times 10^{-4}$.

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* On leave from Institute of Atomic Energy, China

II-A-3

Scattering of 14.9 and 18.0 MeV neutrons from ^{118}Sn

Satoshi Chiba, Yoshimaro Yamanouti, Masayoshi Sugimoto, Yutaka Furuta, Motoharu Mizumoto, Mikio Hyakutake*, and Shin Iwasaki**

Using the JAERI tandem fast neutron time-of-flight(TOF) spectrometer, neutron elastic and inelastic scattering cross sections to low lying states of ^{118}Sn have been measured at incident neutron energies of 14.9 and 18.0 MeV.

Neutrons were produced by the $D(d,n)^3He$ reaction. The deuteron beam was extracted from the in-terminal ion source. The scatterer was a metallic cylinder of 1.6cm in dia. and 3cm in height, 97.2% in ^{118}Sn. Scattered neutrons were detected by an array of four 22cm dia. and 35cm thick NE213 liquid scintillators. The flight path was 8m long.

The elastic and inelastic scattering cross sections to the first 2^+ (Q=-1.23 MeV) and the 3^- group (Q=-2.32 MeV) were obtained. After correcting the effects of finite sample size, the present data will be analyzed by the spherical optical model and DWBA.

* Kyushu University
** Tohoku University

B. Nuclear Data Center, Department of Physics and Working Groups of Japanese Nuclear Data Committee

II-B-1 Evaluation of Heavy Nuclide Data for JENDL-3

JNDC Subworking Group on Heavy Nuclide Data

Evaluation work on neutron nuclear data of Th, Pa, U, Np and Pu isotopes is in progress for JENDL-3. The fission cross sections of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu, and the capture cross sections of ^{238}U and ^{197}Au are evaluated by means of the simultaneous evaluation method, in the energy range between 50 keV and 20 MeV. The DWBA and coupled channel calculations are widely applied for the evaluation of the inelastic scattering cross sections. As many recent experimental data as possible will be adopted for resonance parameters and cross sections.

II-B-2

Evaluation of Neutron Nuclear Data for ^{250}Cf and ^{251}Cf

Tsuneo NAKAGAWA

A paper on this subject was published as JAERI-M 86-086 with the following abstract:

Nuclear data of ^{250}Cf and ^{251}Cf have been evaluated in the energy range from 10^{-5} eV to 20 MeV. There exist only a few experimental data for the cross sections of both the isotopes, that is, the cross sections at the thermal neutron energy and the resonance integrals. Therefore, the present evaluation was based on systematic trends of the data and the calculation with the optical, statistical and evaporation models. Cross sections evaluated in this work are the total, elastic and inelastic scattering, fission, radiative capture, (n,2n), (n,3n) and (n,4n) reaction cross sections. In the energy range below 150 eV, hypothetical resonance levels were generated so as to reproduce the measured thermal cross sections and resonance integrals. Other evaluated quantities are the angular distributions of elastically and inelastically scattered neutrons and those of neutrons emitted from the (n,2n), (n,3n), (n,4n) and fission reactions, their energy distributions, and average numbers of neutrons emitted per fission. The results were compiled in the ENDF/B-V format.

II-B-3

Activity of Decay Heat Evaluation Working Group

Decay Heat Evaluation Working Group

Although the first version of the JNDC FP Decay Data File reproduced the measured decay-heat remarkably well, some discrepancies still remained.[1,2] In the course of an attempt for revision of the file, where the decay schemes of influential nuclides were reexamined in detail, the sources of the above discrepancies were identified fairly well.[3] By replacing the average β-and γ-ray energies with the new data for these nuclides we attained a clear improvement in the decay heat calculations. Examples are given for the case of U-235 (Figs. 1 and 2). These data for the average energies will constitute an essential part of the JNDC FP Decay Data File Version 2, which is planned to be completed within the fiscal year 1986.

As a part of the Japan-US cooperation in the decay-heat field, the Japanese average energies were incorpolated into the American file ENDF/B-V and applied to decay-heat calculation by the Los Alamos group[4]. A part of the results is shown in Figs. 3 and 4. From the comparisons shown here it is easy to assume that the essential difference between the Japanese and the US files comes from the data for average β- and γ-energies.

A method was proposed for estimating the energy spectra of the γ-rays which follow a high Q-value β-decay of short-lived FP nuclides.[5] The method was applied to the calculation of the delayed γ-ray spectra for FPs which lack experimental spectra. These theoretical spectra well complement the summation calculation of the γ-ray spectrum from an irradiated fissile sample after a short cooling time (Fig. 5)

Further an approximate method to describe the neutron capture effect on the FP decay heat was proposed, which was essentially based on a careful simplification of the capture chains of the FP nuclides.[6]

References:

1) T.Yoshida and R.Nakashima, J.Nucl.Sci.Technol., 18, (1981), 393
2) K.Tasaka et al., "JNDC Nuclear Data Library of Fission Products," JAERI-1287 (1983)
3) J.Katakura and R.Nakasima, "Reevaluation of Decay Energies of Fission Product Nuclides" JAERI-M 86-041 (1986)
4) T.R.England, private communication (1986)
5) T.Yoshida and J.Katakura, Nucl. Sci. Engn., 93, (1986), 193
6) K.Tasaka and S.Iijima, J. Nucl. Sci. Technol, to be published
7) M.Akiyama, in preparation for Publication

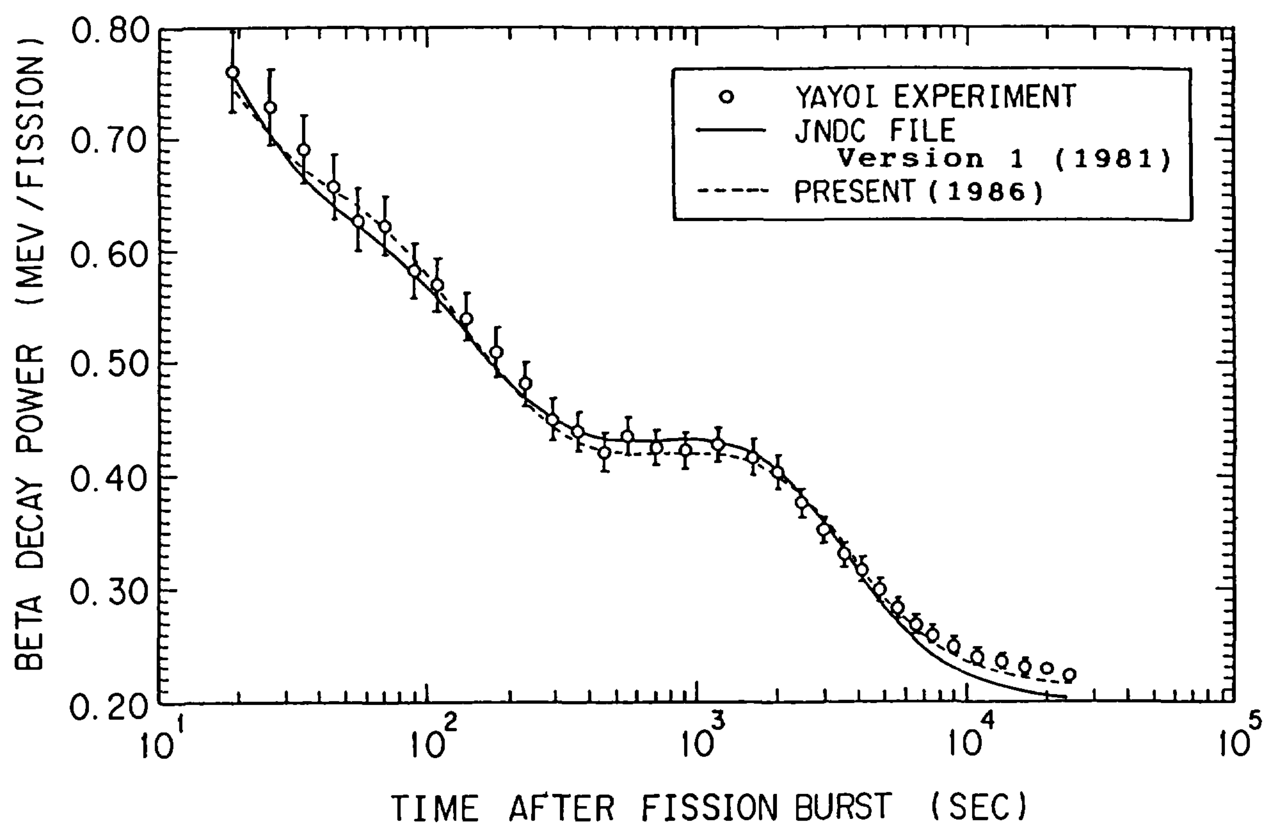

Fig.1 U-235 Decay Heat After Fast Fission Burst (Beta-Ray Component)

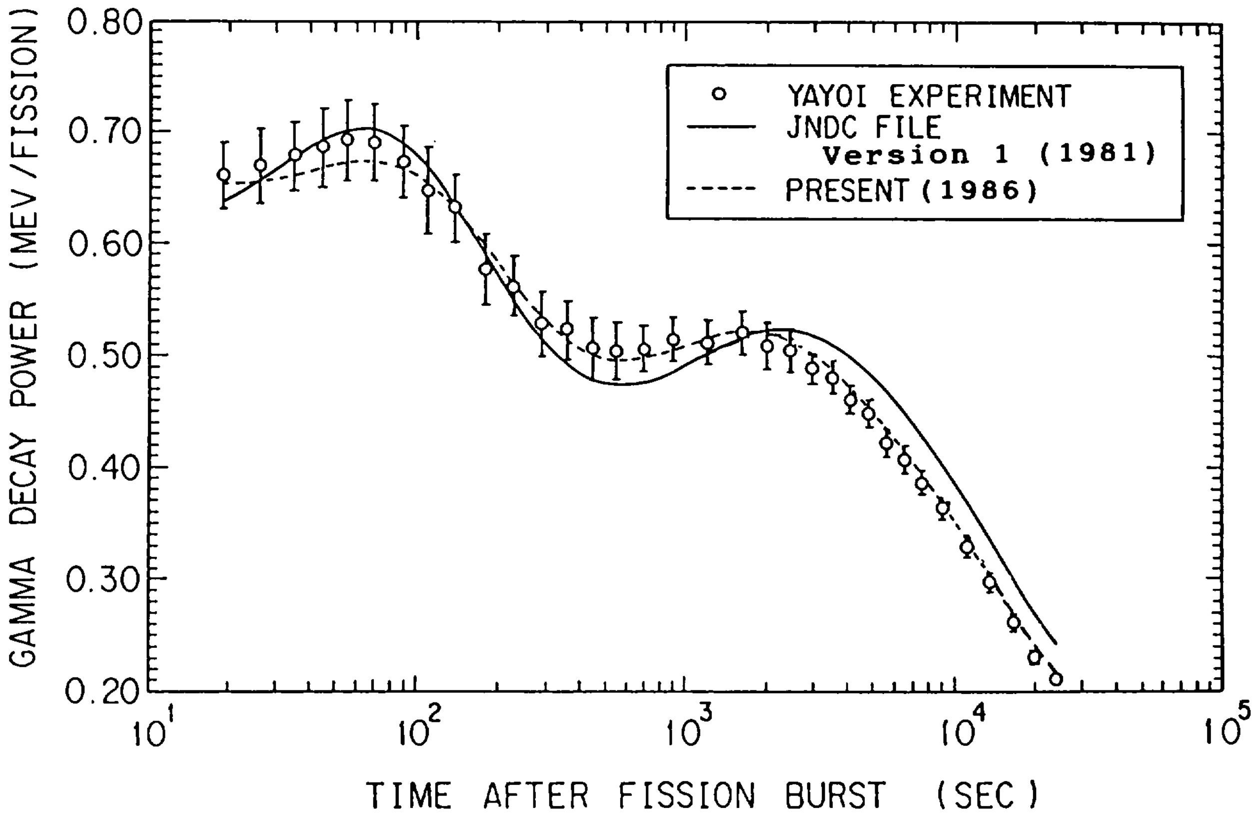

Fig.2 U-235 Decay Heat After Fast Fission Burst (Gamma-Ray Component)

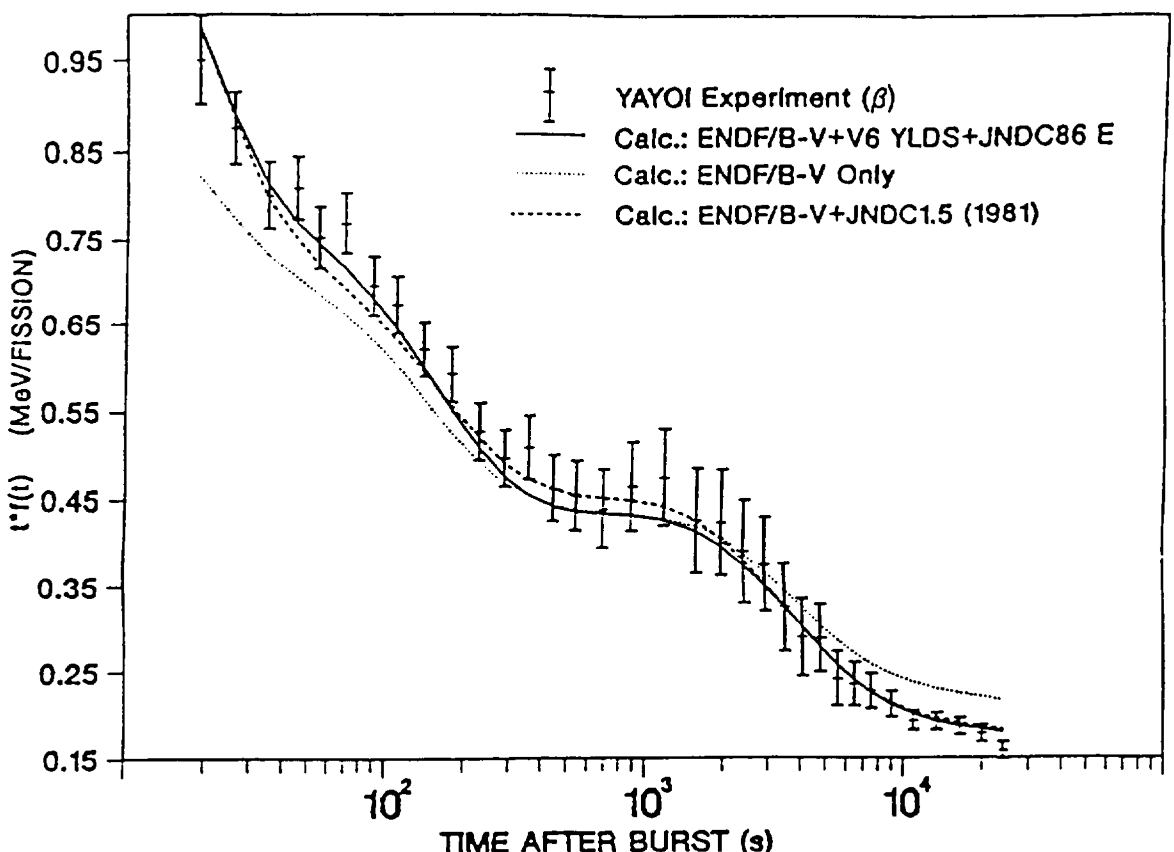

Fig.3. BETA DECAY POWER AFTER ^{238}U FAST FISSION (PULSE)

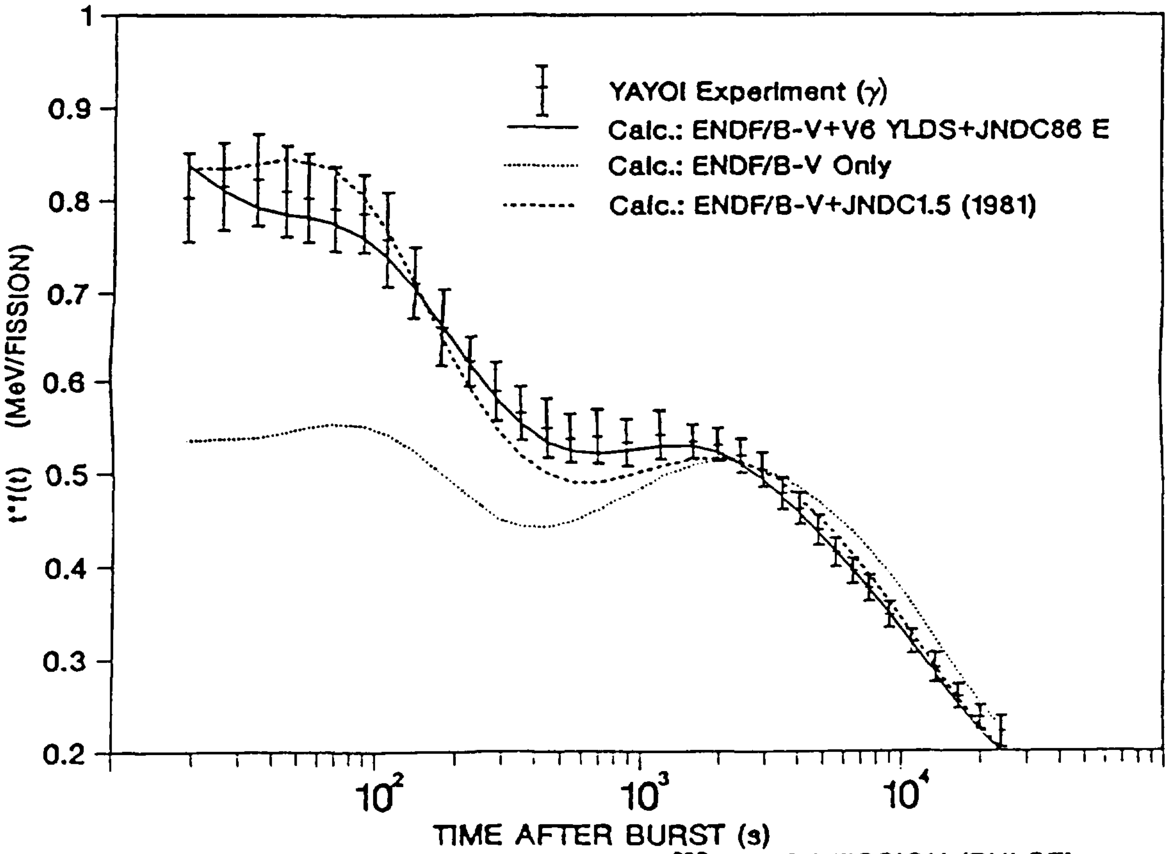

Fig.4. GAMMA DECAY POWER AFTER ^{238}U FAST FISSION (PULSE)

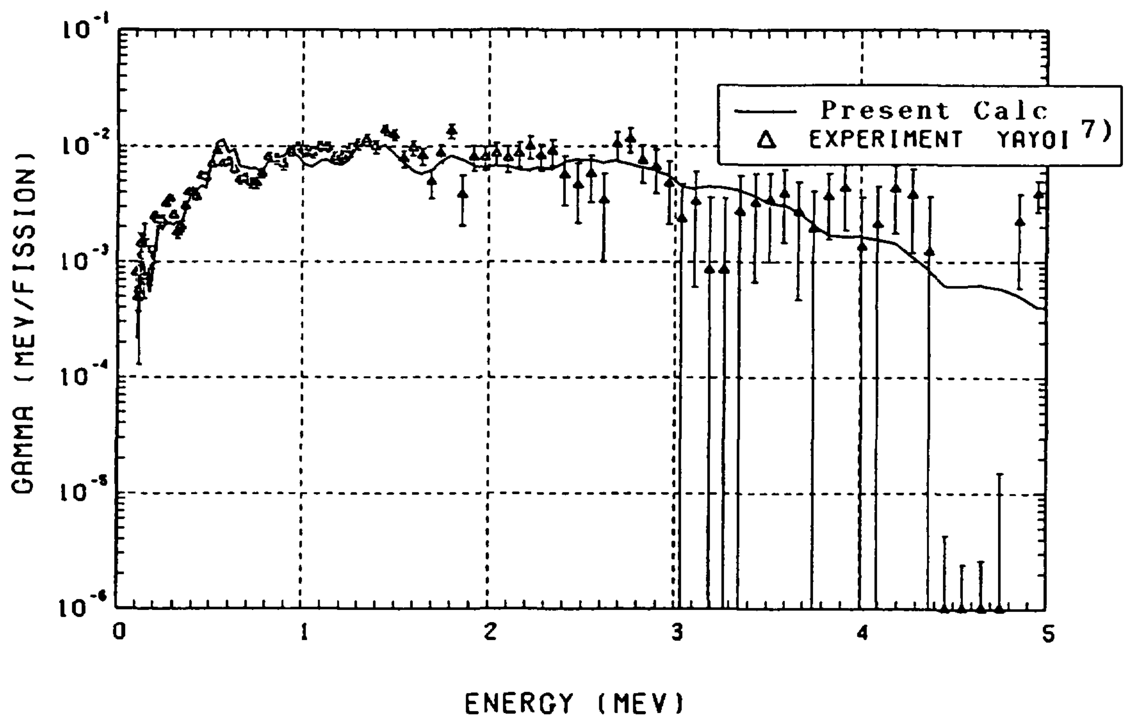

Fig.5 Gamma-Ray Energy Spectrum After 19 Sec Cooling (Pu-239 Fast Fission)

C. Fusion Reactor Physics Laboratory

Department of Reactor Engineering

II-C-1

Meaurement of $^{7}Li(n,n'\alpha)^{3}T$ Cross Section Between 13.3 and 14.9 MeV[+]

H. Maekawa, K. Tsuda, Y. Ikeda, K. Oishi[*1] and T. Iguchi[*2]

From the result of simulated fusion blanket experiment, it was pointed out that a calculation using ENDF/B-IV overestimated the tritium production rates by about 15 %.(1) There are large differences among recently measured data. For example, the data of H. Liskien et al.(2) is lower than those of ENDF/B-IV by (8.6 - 18.5) %, those of D. L. Smith et al.(3) by (5 - 20) % and those of M. T. Swinhoe(4) by about 25 %. Therefore, we have measured the cross section of $^{7}Li(n,n'\alpha)^{3}T$ accurately using the new technique of tritium production

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\+ This content is also published in Reactor Engineering Department Annual Report (Apr. 1, 1985 - Mar. 31, 1986), JAERI.

*1 Visit Scientist from Shimizu Construction Co. Ltd.

*2 Faculty of Engineering, The University of Tokyo.

*3 Collaborative Program between JAERI and The University of Tokyo.

rate measurement developed recently at JAERI.[5] At the same time, Dierckx's method [6] was also applied to the experiment by the University of Tokyo group.*3

Irradiated samples of JAERI were sintered $^{7}Li_2O$ pellets of 12 mm dia. × 2 mm and 85 % T.D. Those of the University of Tokyo were cold-pressed $^{7}Li_2CO_3$ pellets of 12 mm dia. × 3.5 mm. Starting material of both pellets was the same enriched $^{7}Li_2CO_3$ powder (^{7}Li : 99.926 ± 0.005 atom %). These samples were sandwiched by aluminum and niobium foils, which were used as neutron flux monitors. They were placed surrounding the target. The distance from the target to the each sample was about 10 cm. Irradiation time, total D-T neutron yield and average yield rate were 22500 s (∿ 6 h), 4.46×10^{15} and 1.98×10^{11}, respectively.

Irradiated pellets of both groups were treated chemically and the amount of produced tritium was measured by liquid sintillation counting method. In the cases of $^{7}Li_2O$ pellets, some of produced tritium in the pellets was unresolved and remained in the gas-phase during the resolving procedure by water. The tritium in the gas-phase was trapped and measured separately from that in the liquid-phase. The fraction of escaped tritons during the irradiation was estimated by the independent experiment. Then we got total amount of produced tritium in the pellets accurately.

Measured cross section data by $^{7}Li_2O$ pellets are shown in Fig. 1 with JENDL-3PR1, ENDF/B-IV and -V curves. These data are based on the cross section of $^{27}Al(n,\alpha)^{24}Na$ as the standard. Neutron energy at the each sample position was estimated by use of source spectrum calculated by the Monte Carlo code MORSE-DD. Systematic and accidental errors are 4.1 and 2.3 %, respectively. Measured data by $^{7}Li_2CO_3$ pellets are also shown in Fig. 1. In this case, the cross section of $^{93}Nb(n,2n)^{92}Nb$ is used as the standard.

Good agreement has been obtained between the results of JAERI ($^{7}Li_2O$) and those of the University of Tokyo ($^{7}Li_2CO_3$). They also agree with those of Liskien et al.[(2)] and Osaka University.[(7)] The data of JENDL-3PR1 and -3PR2 are about 7 % lower than the present results.

References

(1) Maekawa H., et al. : JAERI-M 83-196 (1983).

(2) Liskien H., et al. : Int Conf on Nuclear Data, Antwerp (1982).

(3) Smith D. L., et al. : ANL-NDM-87 (1984).

(4) Swinhoe M., et al. : Nucl. Sci. Eng., 89, 261 (1985).

(5) Tsuda K., Maekawa H. : To be published.

(6) Dierckx R, : Nucl. Instr. Meth., 107, 397 (1973)

(7) Takahashi A., et al. : Private communication.

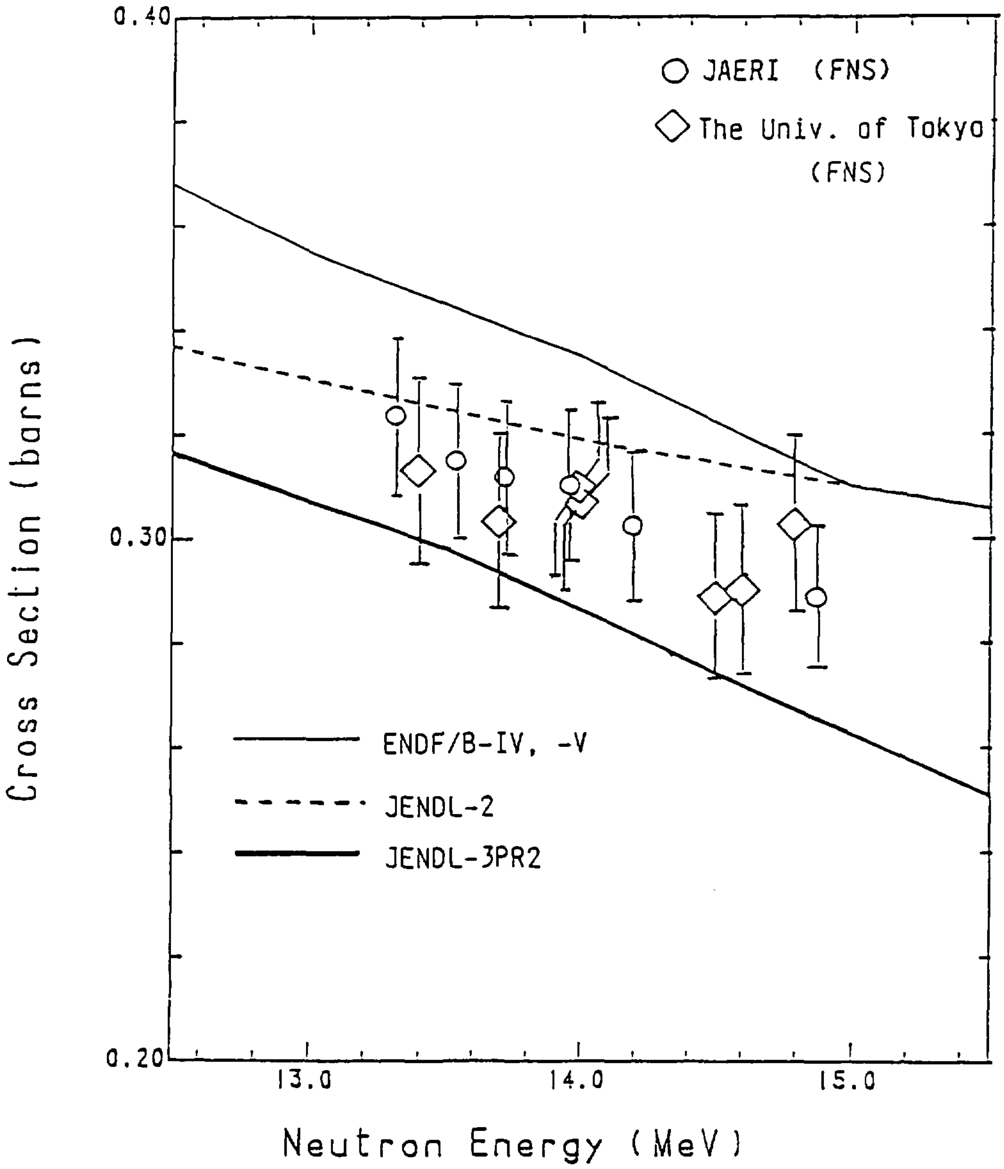

Fig. 1 Measured and evaluated cross section of $^{7}Li(n,n'\alpha)^{3}T$

II-C-2

Activation Cross Section Measurement of Zr and Cr for 14 MeV Neutrons+

Y. Ikeda, C. Konno, K. Oishi**, T. Nakamura, H. Miyade*, K. Kawade*, H. Yamamoto*, and T. Katoh*

The method and irradiation configurations were the same as the former case reported in the annual report in 1985.(1) The measured reactions and associated nuclear data are given in Table 1.

As mentioned in the previous report there were several adjacent isotopes in Zr and Cr which make it difficult to deduce the actual contribution of the interest reaction from other ones by competing reaction. The measured cross sections were corrected by the abundance of the target nuclei in the sample.

In usual cases, neutron intensity at the D-T target was about 1.5 $\times$ 10^{11}n/sec. The detector efficiency of the Ge detectors was calibrated using a set of calibrated gamma-ray sources and the directly irradiated sample.

The obtained cross section for the ^{52}Cr(n,2n), ^{90}Zr(n,p), ^{91}Zr(n,p), and ^{91}Zr(n,np) are shown in Figs. 1 to 4, respectively, with the data measured by other persons and evaluated in JENDL-2 and ENDF. Present data are smooth over the energy and cover a wider energy

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+ This content is also published in Reactor Engineering Departtment Annual Report (Apr.1, 1986 - Mar. 31, 1986), JAERI.

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range. The experimental errors are smaller than those of other data.

We have measured the cross sections on more than fifteen elements and more than fifty reactions and now planning to extend the number of the data on the Ni, W and Sn isotopes. Those data measured are expected to be useful in the next activation cross section library such as JENDL and ENDF.

Reference

(1) Ikeda Y., et al. : JAERI-M 85-116 (1985) 109

Table 1 Reactions and the associated decay parameters

Target Nucleus	Reaction	Product	Half-life	Gamma-ray Energy(keV)	Branching Ratio(%)
^{50}Cr	(n,2n)	^{49}Cr	41.9m	152.9	29.1
^{52}Cr	(n,2n)	^{51}Cr	27.7d	320.1	10.2
^{90}Zr	(n,α)	^{87m}Sr	2.81h	388.4	87.0
	(n,p)	^{90m}Y	3.19h	479.5	91.0
^{91}Zr	(n,np)	^{90m}Y	3.19h	479.5	91.0
	(n,p)	^{91m}Y	49.7m	555.6	94.9
^{92}Zr	(n,np)	^{91m}Y	49.7m	555.6	94.9
	(n,p)	^{92}Y	3.54h	934.5	13.9
^{94}Zr	(n,np)	^{93}Y	10.25d	266.9	6.8
	(n,α)	^{91}Sr	9.48h	1024.3	33.0
^{96}Zr	(n,2n)	^{95}Zr	63.98d	756.7	54.6

*Data were taken from Table of Isotopes, 7th Edition

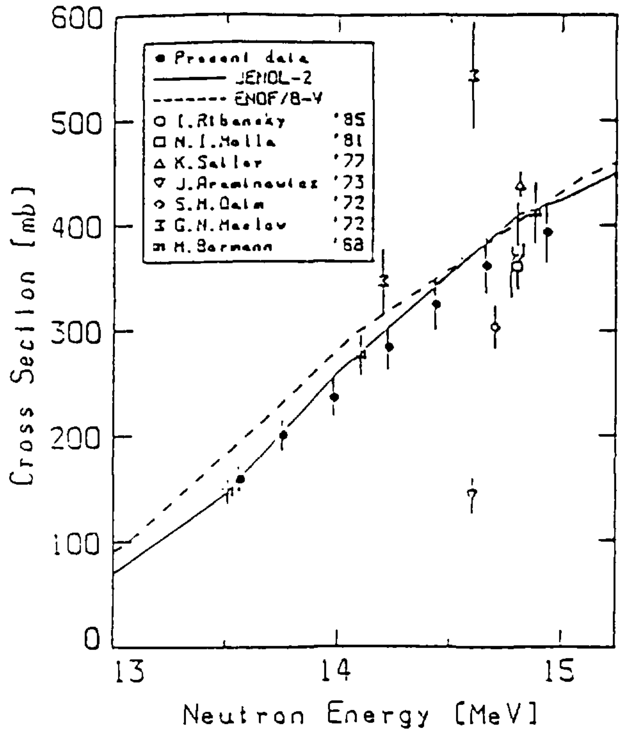

Fig. 1 $^{52}Cr(n,2n)^{51}Cr$

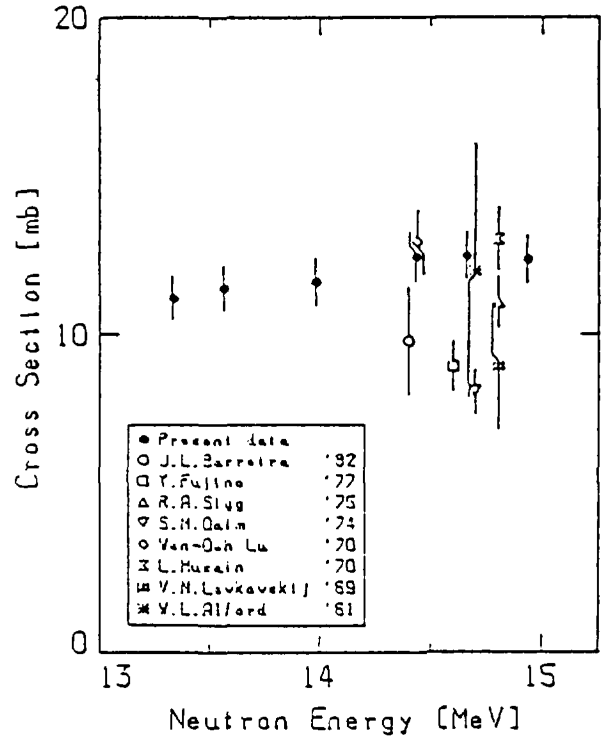

Fig. 2 $^{90}Zr(n,p)^{90m}Y$

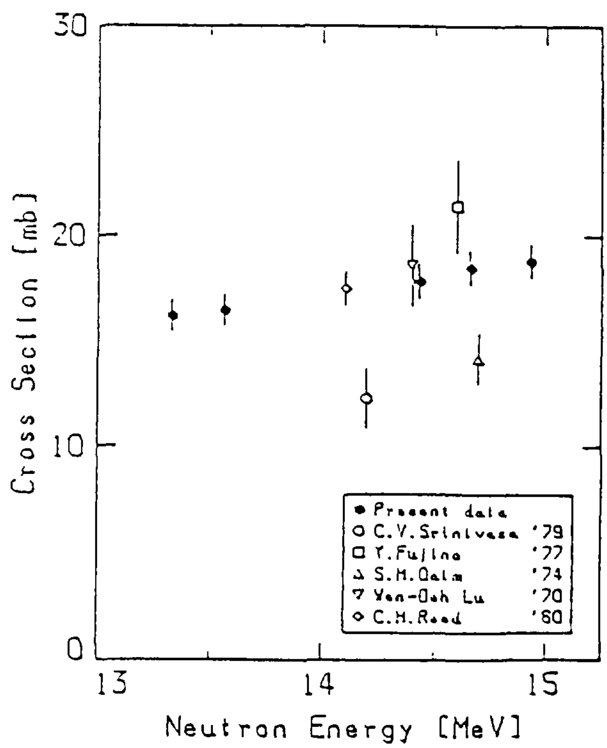

Fig. 3 $^{91}Zr(n,p)^{91m}Y$

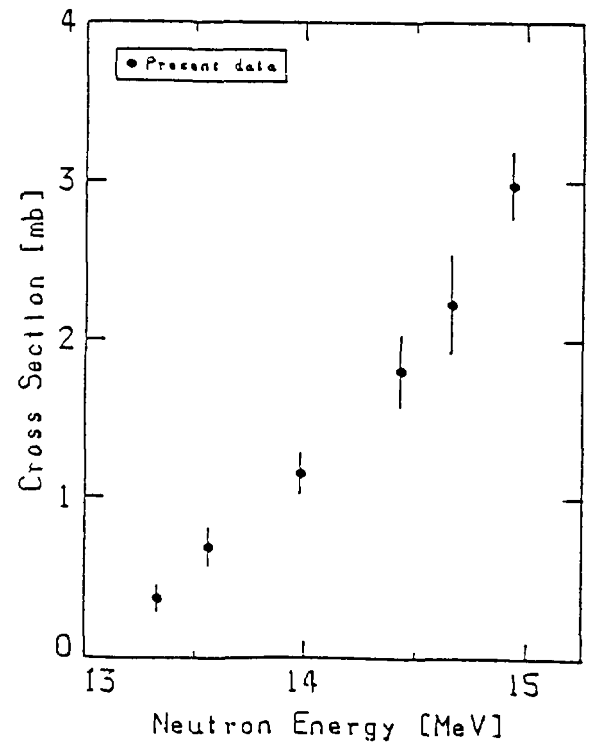

Fig. 4 $^{91}Zr(n,np)^{90m}Y$

II-C-3

Masurements of Activation Cross Sections+

—— Short-Lived Nuclides ——

C. Konno, Y. Ikeda, K. Oishi**, H. Maekawa, T. Nakamura,

T. Yamada*, K. Kawade*, H. Yamamoto*, and T. Katoh*

The effects of short-lived radioactive nuclei are important for dose estimation after shutdown of the D-T fusion reactor. Systematic measurements on short-lived activation cross sections were planned for several elements included in the candidates of structural materials for the fusion reactor. This year the cross sections of the reactions shown in Table 1 have been measured.

Samples were irradiated by D-T neutrons at the end of the 80° beam line of the FNS facility. Since the produced nuclei are short-lived, six temporary pneumatic tubes have been set from the target to the outside of the room. The end of each tube in the target room was placed around the Ti-T target, at the angles of 5°, 45°, 65°, 95°, 135°, and 165°, to the d^+ beam, so as to change the incident neutron energy. The irradiation time was 2 ∿ 10 minutes. Gamma rays emitted from samples were measured using a Ge detector immediately after samples were returned to the outside through the pneumatic tubes. The neutron flux was estimated by reaction rate of the aluminum foil which was irradiated separately at the sample position through α-monitor

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+ This content is also published in Reactor Engineering Department Annual Report (Apr. 1, 1985 - Mar.31, 1986), JAERI.

* Nagoya University.

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counts as a relative monitor in both reaction measurements. The data of ENDF/B-V was used as $^{27}Al(n,\alpha)^{24}Na$ cross section. Obtained neutron fluxes were $1.2 \times 10^8 \sim 2.5 \times 10^7$ n/cm^2/sec at each sample position. The total errors of cross sections using above method were within ± 10 % except the rare reactions, e.g. $^{94}Zr(n,p)^{94}Y$, $^{94}Mo(n,p)^{94m}Nb$, $^{97}Mo(n,p)^{97m}Nb$, and $^{98}Mo(n,np)^{97m}Nb$.

The obtained results for $^{51}V(n,p)^{51}Ti$ and $^{92}Mo(n,\alpha)^{89m}Zr$ were shown in Figs. 1 and 2, respectively.

Table 1 Measured reactions

Reaction	Half-life	Eγ(keV)	Branching Ratio(%)
$^{25}Mg(n,p)^{25}Na$	60 s	975.2	14.7
$^{27}Al(n,p)^{27}Mg$	9.46 m	843.8	73
$^{28}Si(n,p)^{28}Al$	2.24 m	1778.7	100
$^{29}Si(n,p)^{29}Al$	5.56 m	1273.2	89.1
$^{30}Si(n,\alpha)^{27}Mg$	9.462 m	843.8	73
$^{51}V(n,p)^{51}Ti$	5.8 m	319.7	93.4
$^{52}Cr(n,p)^{52}V$	3.746 m	1434.1	100
$^{63}Cu(n,2n)^{62}Cu$	9.73 m	511.0	209.2
$^{90}Zr(n,2n)^{89m}Zr$	4.18 m	587.7	89.5
$^{94}Zr(n,p)^{94}Y$	18.7 m	918.8	56
$^{92}Mo(n,\alpha)^{89m}Zr$	4.18 m	587.8	89.5
$^{92}Mo(n,2n)^{91m}Mo$	64 s	652.9	48.1
$^{94}Mo(n,p)^{94m}Nb$	6.26 m	871	0.47
$^{97}Mo(n,p)^{97m}Nb$	60 s	747	97.9
$^{98}Mo(n,np)^{97m}Nb$	60 s	747	97.9

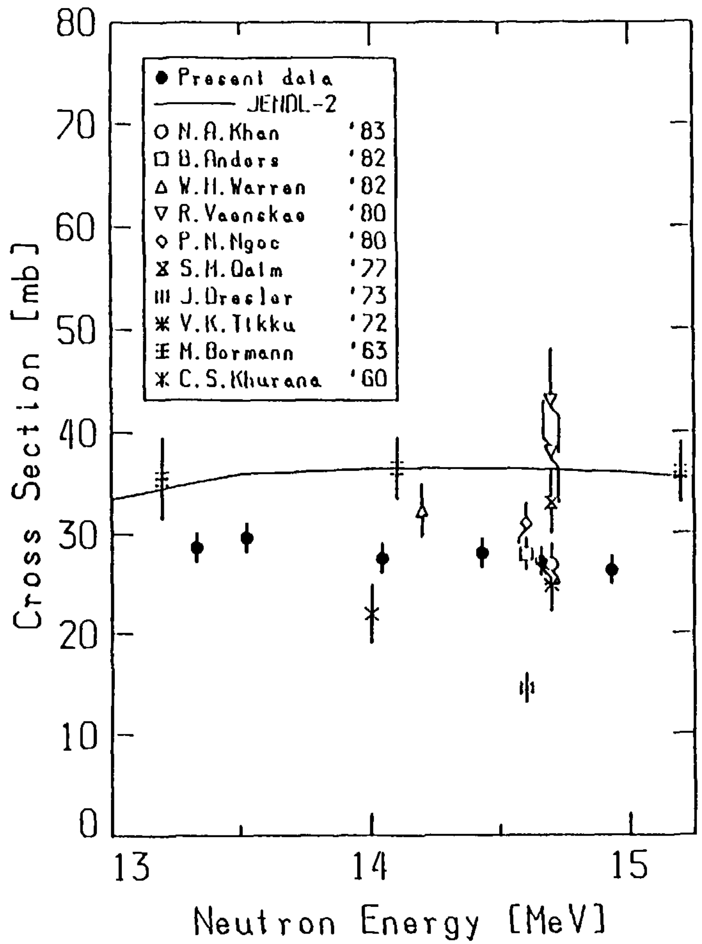

Fig. 1 Cross section of $^{51}V(n,p)^{51}Ti$

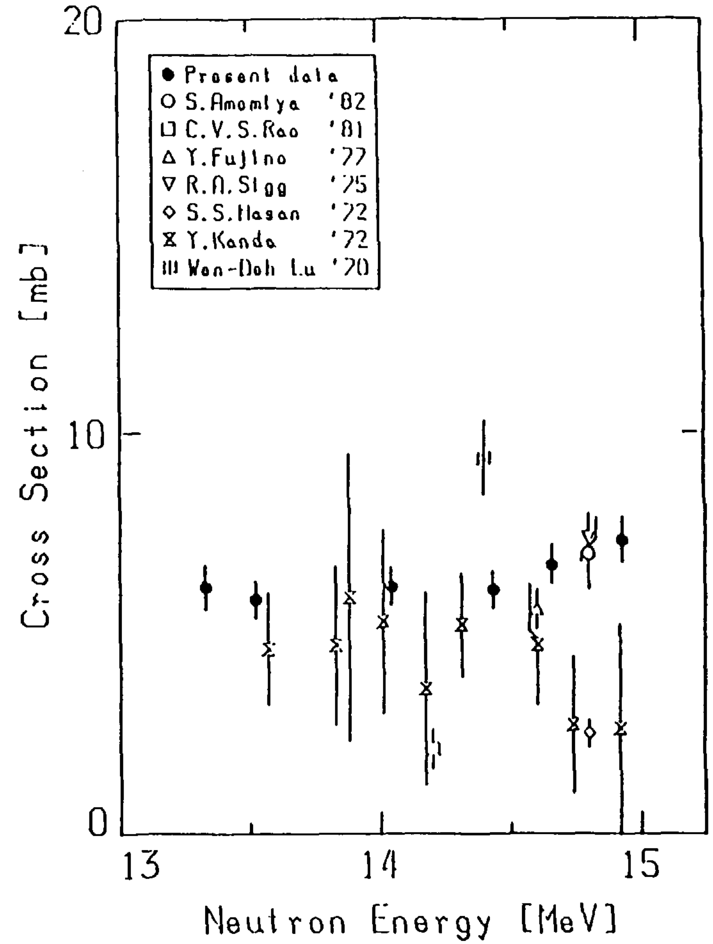

Fig. 2 Cross section of $^{92}Mo(n,\alpha)^{89m}Zr$

II-C-4

Experiment and Analysis of Induced Activities in Concrete Irradiated by 14-MeV Neutrons[+]

K. Oishi[*], Y. Ikeda, C. Konno, H. Maekawa, and T. Nakamura

The deuterium-tritium neutrons were generated in the Ti-T rotating target placed at the end of the 0-deg beam line of the FNS facility. Eight samples of the concrete aggregates were selected, and emplaced at the distance of $\sim$ 25 cm from the target in the direction of 45 deg to the d^+ beam line. Niobium foils were used to monitor the neutron flux for each sample. The irradiation time was $\sim$ 43 h, and the measured neutron yield was 3.4×10^{17}. From one day to half a year, after irradiation gamma-rays emitted from the produced activities were measured by Germanium detectors. Reaction rates were deduced from the gamma-ray counts with necessary corrections.

For this irradiation condition, it was proved that ^{42}K, ^{24}Na, ^{43}K, ^{48}Sc, ^{47}Sc, ^{47}Ca, ^{46}Sc, and ^{54}Mn, in half life order, make an important contribution to the total activity. In addition, the comparison between the experiment and calculation was made. The calculations were performed using the fusion radioactivity calculation code THIDA [(1)] and its related library CROSSLIB with the input neutron flux generated from the one-dimensional transport code ANISN and the nuclear library GICX-40.[(2)] For ^{24}Na and ^{54}Mn, whose cross sections

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+ The content of this paper is submitted to the 7th ANS Topical Meeting on Technology of Fusion Energy, June 16-19, 1986, Reno, Nevada.

* Visiting Scientist from Shimizu Construction Co., Ltd.

were well estimated, agreement between the experiment and calculation was within ± 10 %, which proved the validity of the calculational code. For reaction rates caused by Calcium and Titanium isotopes, however, the calculational results differed from the experimental ones between - 20 % to + 40 % were obtained.(see Fig. 1) This inconsistency was caused by the uncertainty of cross sections around 14 MeV, because the incident neutron energy was almost 14 MeV.

Cross section measurements around 14 MeV were performed at FNS and listed in Table 1. Samples, except ^{48}Ca, were separated isotopes; the abundance of remarkable nuclide was > 90 %. The samples positioned at 10 cm in the radius, centered the target from 2.8 to 165 deg to the d^{+} beam. The incident neutron energy was from 13.3 to 14.9 MeV. The neutron fluxes were monitored using Niobium foils. The neutron source intensity was $\sim$ 2 × 10^{11} n/s at the target. Consequently, calculations were performed again using the measured cross sections (see Table 2), and then the agreement between experiment and calculation was improved within ± 10 %. (see Fig. 1)

References

(1) Iida H. and Igarashi M. : JAERI-M 8019 (1978)

(2) Seki Y., et al. :JAERI-M 8818 (1980)

Table 1 Nuclear parameters for reactions to be measured

Reaction	$T_{1/2}$	Eγ(MeV)	Iγ(%)
$^{42}Ca(n,p)^{42}K$	12.36 h	1.5247	17.9
$^{43}Ca(n,p)^{43}K$	22.4 h	0.3729 0.6178	86.7 80.0
$^{44}Ca(n,np)^{43}K$	"	"	"
$^{48}Ca(n,2n)^{47}Ca$	4.536 d	1.2971	76.0
$^{46}Ti(n,p)^{46g}Sc$	83.8 d	0.88925 1.12051	100.0 100.0
$^{47}Ti(n,np)^{46g}Sc$	"	"	"
$^{47}Ti(n,p)^{47}Sc$	3.35 d	0.1594	68.0
$^{48}Ti(n.np)^{47}Sc$	"	"	"
$^{48}Ti(n,p)^{48}Sc$	43.7 h	0.9835 1.03750 1.31209	100.0 97.5 100.0
$^{49}Ti(n,np)^{48}Sc$	"	"	"
$^{50}Ti(n,\alpha)^{47}Ca$	4.536 d	1.2971	76.0

Table 2 Comparison between cross sections of CROSSLIB and those of present measurement

Reaction	CROSSLIB σ(mb)	PRESENT σ(mb)	abundance (%)	CROSSLIB / PRESENT
$^{42}Ca(n,p)^{42}K$ $^{43}Ca(n,np)^{42}K$	152.0 30.0	116.0 ——	0.65 0.145	0.92
$^{43}Ca(n,p)^{43}K$ $^{44}Ca(n,np)^{43}K$	154.0 2.6	103.0 2.6	0.145 2.09	1.36
$^{46}Ti(n,p)^{46g}Sc$ $^{47}Ti(n,np)^{46g}Sc$	258.36 17.60	229.78 66.47	7.99 7.32	0.94
$^{47}Ti(n,p)^{47}Sc$ $^{48}Ti(n,np)^{47}Sc$	112.34 10.60	128.60 15.74	7.32 73.99	0.76
$^{48}Ti(n,p)^{48}Sc$ $^{49}Ti(n,np)^{48}Sc$	65.40 6.80	59.32 6.78	73.99 5.46	1.10
$^{50}Ti(n,\alpha)^{47}Ca$ $^{48}Ca(n,2n)^{47}Ca$	9.00 800.00	9.39 843.74	5.25 32.43*	0.95

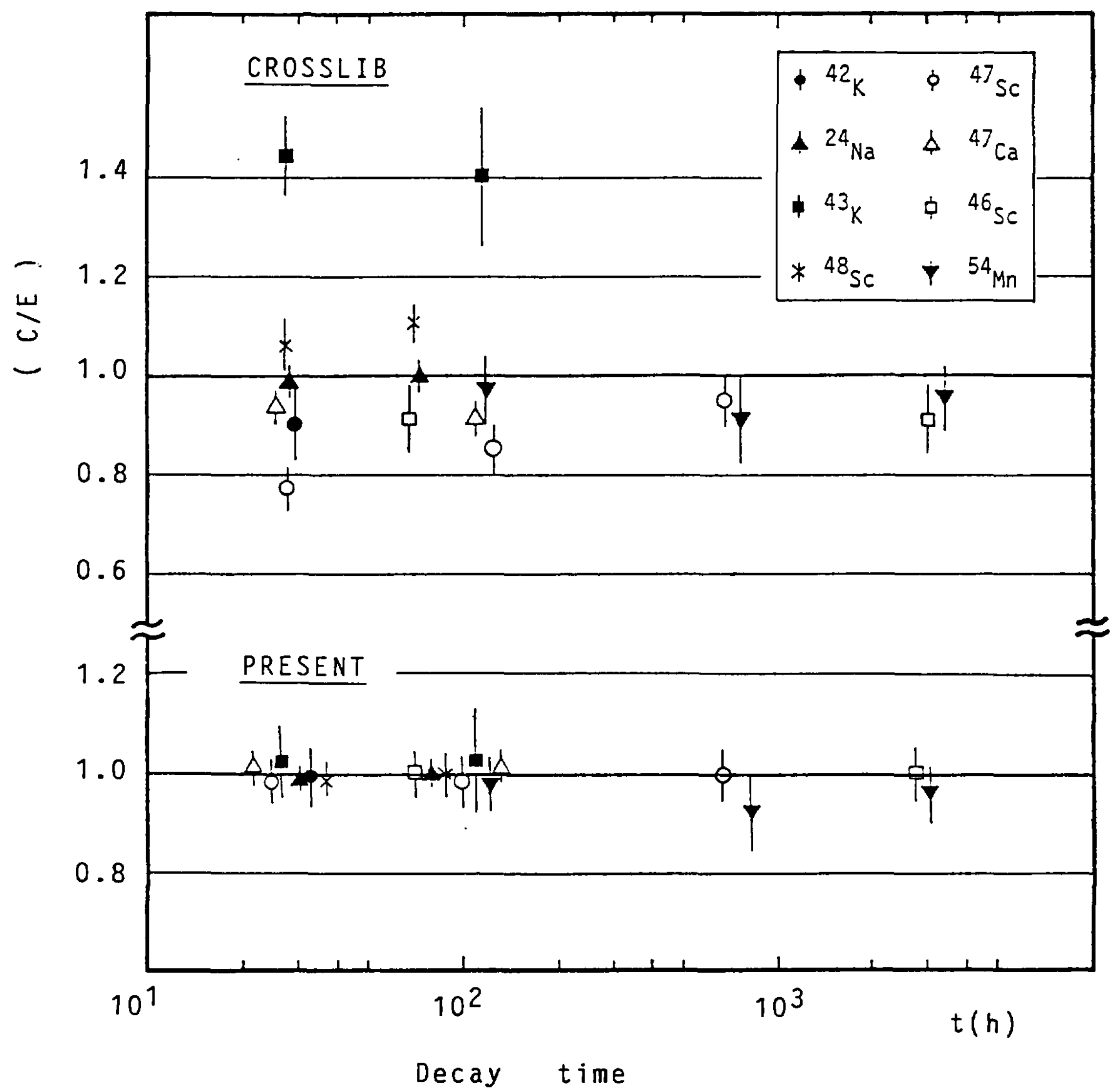

Fig. 1 The calculation-to-experimental (C/E) value of gamma ray intensity. Calculations were performed with the input cross section of CROSSLIB and present data, respectively.

II-C-5

Measurement and Analysis of Angle-Dependent Neutron Spectra Leaking from Beryllium Slab Assembly[+]

Yukio OYAMA, Hiroshi MAEKAWA and Jungchung JUNG[*]

A paper on this subject will be submitted to Nucl. Sci. Eng.

Angle-dependent neutron spectra leaking from beyllium slab have been measured by the time-of-flight (TOF) method. The experimental arrangement was the same as the previous work.[1] The experiment was analyzed by using the DOT3.5 code with the JENDL-3PR1 and ENDF/B-IV data. This experiment was also analyzed by using the MCNP code with the ENDF/B-V data.

The calculated result using each nuclear data file is shown in Fig. 1 with the experimental result. There exist large differences among the nuclear data files used.

Reference

1) Y. Oyama and H. Maekawa, JAERI-M 83-195 (1983), Nucl. Instr. Meth A245 173 (1986)

+ This work is a part of JAERI-USDOE collaboratve program on fusion blanket neutronics.

* Argonne National Laboratory.

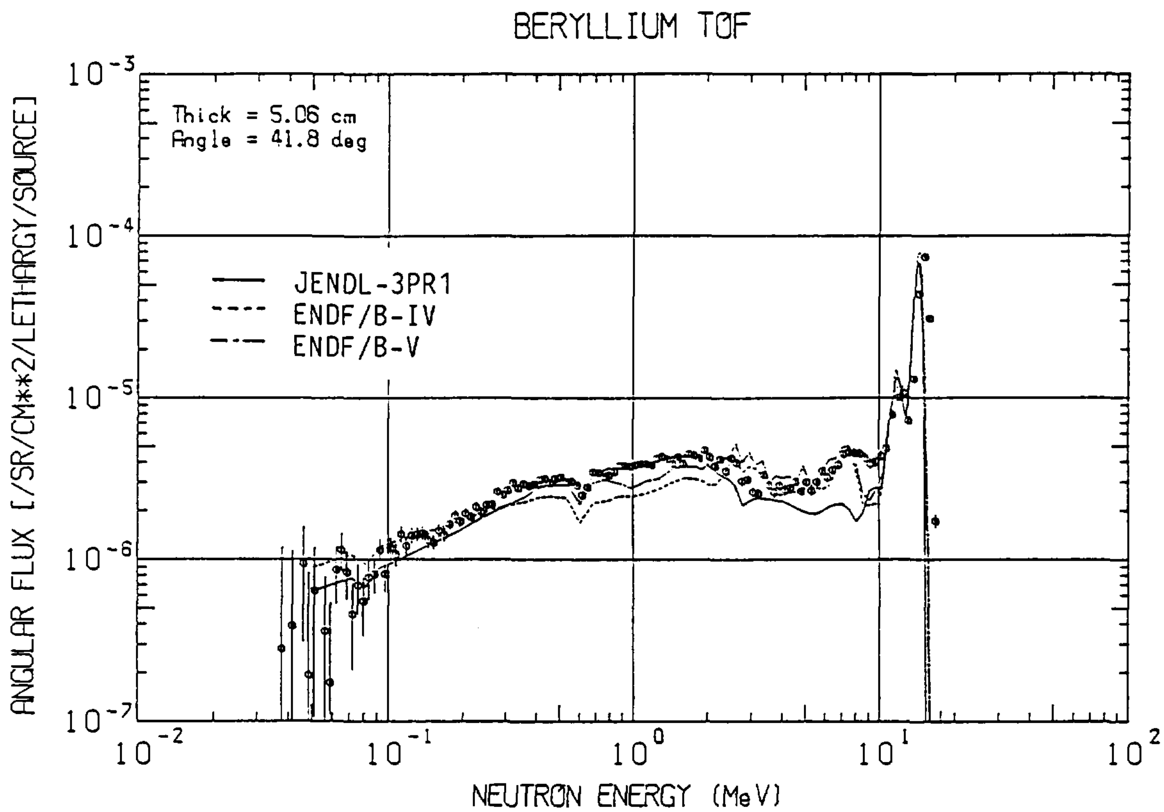

Fig. 1 Measured and calculated results of angular flux on the rear surface of Be

Neutron Cross-Section Sets of 125-group for Fusion Neutronic Calculations+

K. Kosako, H. Maekawa and T. Nakamura

Neutron cross-section sets of 125-group have been prepared at FNS for analysis of blanket benchmark experiments. Backgrounds of this work are as follows:

1) The widely used GICX40 cross-section set(1) has small group number of 42-group neutron and is based on nuclear data of the ENDF/B-IV.
2) The preliminary version of JENDL-3 for fusion neutronics is produced.
3) New cross-section sets are required for analysing the benchmark experimental data obtained at FNS.

The general features of 125-group cross-section sets provided are shown in Table 1. The group structure conformed to Ref. (2). First steps of the procedure were linear-linear interporation in energy and cross-section, reconstruction of energy dependent neutron cross-section from a combination of resonance parameters and tabulated floor cross-sections, and generation of doppler broadened cross-sections. Second, the group constant was generated from above result with the process code, PROF-GROUCH-G/B.(3) Finally, the group constant for each nucleus was converted into the ANISN format and was collected into a cross-section set. The present cross-section sets were named as

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+ This content is also published in Reactor Engineering Department Annual Report (Apr. 1, 1985 - Mar. 31, 1986), JAERI.

JACKAS, JENGIX, ENFKAS and ENDGIX, respectively, as shown in Table 2.

References

(1) Seki Y., et al.: "Coupled 42-Group Neutron and 21-Group Gamma-ray Cross Section Sets for Fusion Reactor Calculations," JAERI-M 8818, (1980).

(2) Nakagawa M., et al.: "MORSE-DD a Monte Carlo Code using Multi--group Double Differential Form Cross Sections," JAERI-M 84-126, p7-8 (1984).

(3) Hasegawa A.: To be published.

Table 1 The general features of 125-group cross-section sets

items	description
group number	125-group for neutron
Legendre scattering terms	P-5
data format	group independent type cross--section for ANISN code
process code	PROF.GROUCH-G/B
weight functions 1.001E-5 ~ 3.2241E-1 (ev) : maxwell distribution 3.2241E-1 ~ 1.6487E+7(ev) : 1/E or E-flat	
nuclear data files	JENDL-3PR1&2, JENDL-2 ENDF/B-IV&V
nuclide number	13
nuclide name	H-1,Li-6,Li-7,Be-9,C-12, O-16,Na-23,Al-27,Si,Ca,Cr, Fe,Ni

Table 2 125-group cross-section sets

name	weight function		nuclear data files
	thermal*	others**	
JACKAS	Maxwell	E-flat	JENDL2 + JENDL3PR1 & 2
JENGIX	Maxwell	1/E	JENDL2 + JENDL3PR1 & 2
ENFKAS	Maxwell	E-flat	ENDF/B-IV & V
ENDGIX	Maxwell	1/E	ENDF/B-IV & V

* 1.0010E-5 < E < 3.2241E-1 eV ** E > 3.2241E-1 eV

D. Thermal Reactor Physics Laboratory

Department of Reactor Engineering

II-D-1

Evaluation of Delayed Neutron Data for Thermal Fission of ^{235}U Based on Integral Experiments Using SHE

Y. Kaneko, F. Akino and T. Yamane

Evaluation of the delayed neutron data for thermal fission of ^{235}U is attempted through an indirect measurement which is based on comparison between calculation and experiment on the following five integral quantities for Semi-Homogeneous Experiment (SHE):

Data 1 (Inverse kinetic parameter, Λ/β_{eff})

Data 2 (Central reactivity worth of Th, NU and EU rod, $\Delta k_{eff}/\beta_{eff}$)

Data 3 (Central reactivity worth of burnable poison rods, $\Delta k_{eff}/\beta_{eff}$)

Data 4 (Effective multiplication factor, k_{eff})

Data 5 (Increase of prompt neutron decay constant due to insertion of control rod, $\Delta\alpha$)

The SHE facility possesses particular advantages for the present purpose in that it is composed solely of 20 % enriched UO_2 and graphite, both these components having nuclear data that are precisely known and also in that the geometrical simplicity of the cylindrical core shape provides for precise neutronic calculations.

Histogram of the percent deviations of the ratios of calculated to measured values from unity is shown in Fig. 1.

The ratios are obtained with use of the Keepin's delayed neutron data set. The plots for the Data 4 and 5 - not scaled on β_{eff} - are seen to present distributions around their own mean values which are close to zero. In contrast, for Data 1, 2 and 3 - scaled on β_{eff} - their mean values are all definitely positive. The foregoing observations betake that the present nuclear data other than relevant to the delayed neutrons and also the methods adopted for the neutronic calculations are very accurate. Then, the least squares method for an indirect measurement was applied in order to find the most probable values of β_{eff} and decay constants of the precursors λs, assuming that disagreement between calculation and experiment can wholly be attributed to the errors in the delayed neutron data used.

It is concluded that the most probable values of β_{eff} of the various SHE cores should be 4.0 % higher than the values which are calculated with use of the Keepin's delayed neutron data set. This judgement leads to that Keepin's β value 0.0065 ± 0.0005 for thermal fission of ^{235}U should be corrected to 0.00676 ± 0.00011 under the assumption that Keepin's energy spectra $\chi(E)$s do not include so large uncertainties that resulting errors in β_{eff} are negligibly small. This corrected value is a little lower than both values of ENDF/B-IV and V. Experimental error of 1.7 % estimated for the present indirect measurement of β_{eff} is comparable with that of ~6 % which has often been estimated for the direct measurements. On the other hand, most probable values of λs obtained are not away from those for thermal fission filed in the Keepin's delayed neutron data set beyond their experimental uncertainties.

DATA NO.	1	2	3	4	5
INTEGRAL QUANTITY	Λ / β_{eff}	TH, NU, EU ROD WORTH	BP ROD WORTH	EFFECTIVE MULTIPLICATION FACTOR	CONTROL ROD EFFECT $\Delta\alpha$
CORE NAME	SHE - 7, 8 14, B2 T1, 16	SHE - 14	SHE - 8	SHE - 5, 6, 7, 8, 12, 13, 14, 16, T1, T2, B1, B2, B3	SHE - 14

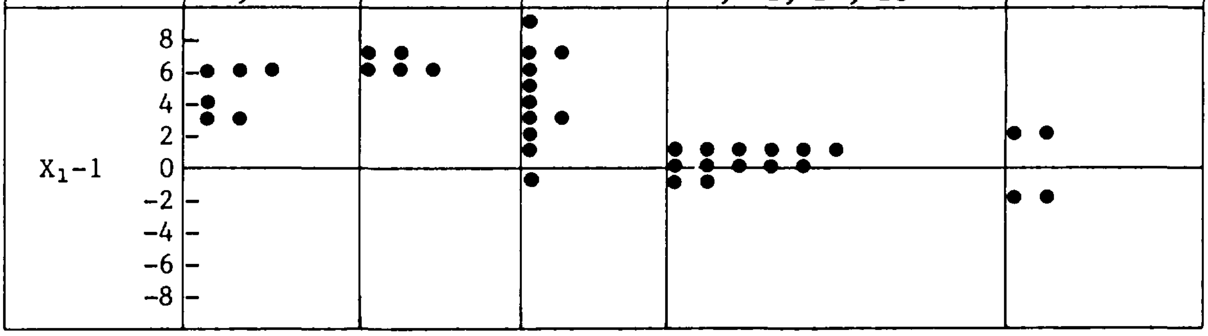

Fig. 1. Histogram of percent deviations of ratios of calculated to measured values for various integral quantities when use is made of Keepin's delayed neutron data.

E. Application and Development Division

Department of Radioisotopes

II-E-1

Measurement of Thermal Neutron Capture Cross Sections of ^{152}Gd and ^{153}Gd

H. Tominaga, T. Imahashi, N. Tachikawa, K. Hoizumi, H. Kato,
A. Sato, and H. Kogure

Data of the (n,γ) reaction cross sections of ^{152}Gd and ^{153}Gd are essential for estimating the yield of ^{153}Gd production, however, show large discrepancies in recent literatures[1)-5)]. In this study, these neutron capture cross sections were measured for reactor thermal neutrons by the method of neutron activation followed by γ-ray spectrometry.

The capture cross section of ^{152}Gd was first determined from the activities of ^{153}Gd produced in samples containing μg quantities of 42.77 % enriched ^{152}Gd by irradiating a small fluence of neutrons ($\sim 6 \times 10^{20}\ m^{-2}$) with the pneumatic tube of the JAERI Research Reactor JRR-2. The value of the cross section obtained without correction for the effects of epithermal neutrons was 599 $\pm$ 14 b. After the correction for the contribution of epithermal neutrons to both the neutron fluence and the produced activities, using a Cd ratio(Au) of $6 \sim 7$, a value of 690 $\pm$ 30 b was obtained.

The capture cross section of ^{153}Gd was also deduced from the ^{153}Gd activities of samples of trace amounts of enriched ^{152}Gd irradiated by a large fluence ($\sim 1 \times 10^{24}\ m^{-2}$) in a in-core hole of the same reactor, by using an equation for the activity yield in the double neutron capture process with an effective value of the capture cross section of ^{152}Gd including the effect of epithermal neutrons (Cd ratio(Au) = 1.5$\sim$2). A value of 32,000 $\pm$ 4,000 b was found as the capture cross section of ^{153}Gd for the reactor thermal neutrons neglecting the unknown resonance integral of ^{153}Gd.

The results obtained are listed in Table 1, compared with previous data. The uncertainties in the data of this work may be reduced considerably by evaluating the effects of epithermal neutrons more accurately.

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Table 1 Present results and comparison with published data

Reference	Method	$\sigma(^{152}Gd)$ (b)	$\sigma(^{153}Gd)$ (b)
Steinnes (1972)	Activation	1,100 ± 100	————
Table of Isotopes(1978)	Compilation	1,100 ± 100	————
Spits (1981)		————	26,000 ± 10,000
Neutron Cross Sections(1984)	Evaluation	735 ± 20	36,000 ± 4,000
Vertebnyi (1984)	Transmission	1,100 ± 230	14,000 ± 3,000
Present work	Activation	690 ± 30	32,000 ± 4,000

III. Kinki University

III-1

A New Empirical Formula for the Prediction of Around 14 MeV Neutron Cross Section on Light Nuclei

O. Horibe and H. Chatani*

(1), basis of the prediction formula. It has become clear from the empirical rule[1] that cross sections σ_0 especially specified by Hughes [2] for out-going charged particles with unit penetrability in the spectrum averaged threshold reaction cross sections can be given by the following equation, respectively for nuclei with the same neutron excess-number.

$$\sigma_0 = A^{2/3}\exp\left[\alpha(E_T + aA) + \beta\right] ,$$

where E_T is threshold reaction energy, A, mass-numbers of target nuclei, a is almost constant for the same neutron excess-number nuclei and α is almost constant or almost insensitive for neutron energies ranging over U-235 fission neutron energies, and β is a constant.

Then, by using σ_0 a prediction formula for the (n,p), (n,α) and here-with for the (n,2n) reaction cross sections, σ, was assumed to be

$$\sigma = \sigma_0 \exp\left[-C(N-Z)/A\right] ,$$

or

$$\sigma = A^{2/3}\exp\left[\alpha(E_T+aA)+\beta\right]\exp\left[-C(N-Z)/A\right]. \qquad (1)$$

In fact, the form of this formula is similar to that of Levkovskii's formula[3] in which an inelastic scattering cross section is replaced by the above σ_0 .

(2), simplification of formula and calculation methods for α , β and C. E_T is given by a reaction Q-value, that is, $-Q(A+1)/A$. Therefore, values of E_T are approximately given by $-Q$ values. Actual values of the constant a for the (n,p) reaction had been known to decrease monotonously from 0.102 MeV for nuclei with (N-Z)=0 to 0.048

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MeV for nuclei with (N-Z)=54 and similarly 0.322 to 0.109 MeV for the (n,α) reaction. Therefore, values for E_T+ aA are roughly given by E_T for small numbers of A. In addition, from the fact that Levkovskii's formula can give successful results in predictions of the cross sections on nuclei of larger mass-numbers than about 30, values for α then expected are to tend to nearly zero. From these considerations, therefore, Eq. (1) was simplified to be

$$\sigma = A^{2/3}\exp(\alpha Q+\beta)\exp[-C(N-Z)/A],$$

or

$$\ln\sigma - (2/3)\ln A + C(N-Z)/A = \alpha Q + \beta. \qquad (2)$$

Consequently, values of the left hand side of the equation must correlate linearly to values of Q. Then, the best fit value of C for Eq. (2) is to give such a best linear correlation as that a standard deviation of actual values of the left hand side of the equation become minimum and the then values of α and β are just the best fit ones to be acceptable as the inclination parameters of the linearly fitted line.

(3), grouping for experimental back data and values of α, β and C. There are many cross section data measured on various nuclei. However, data values for each of most reaction cross sections are rather widely scattering. Therefore, it is difficult to select the best value for each from among such scattering data values. Fortunately, Bormann et al.[4] had surveyed many literature values of the (n,p), (n,α) and (n,2n) reaction cross sections and then recommended typical values for respective cross sections at representative neutron energies of 14.1, 14.4 and 14.9 MeV. Then, these recommended data vaules and additional several data[5,6,7,8,9,10,11] have been used as the prediction back data for the present.

From an actual viewpoint, back data on several different neutron excess-number nuclei were grouped together, so that general differences between predicted and back data values are comparable to those in each separate prediction. The back data thus grouped for the present are presented in Table 1 (a), (b) and (c).

In each grouped data, the best fit value of C was searched by parameter survey varying numerical values of C discretely from 500 to -500 in 10 steps. Values of the fitting parameters thus obtained using Q-values[12] are shown in Table 1 (a), (b) and (c).

(3), estimated values of the cross sections. Using the best fit values of the paramters shown in Table 1, the cross sections have been calculated from Eq.(2). Obtained results are shown in Table 2 (a), (b) and (c) along with the back data values and also for comparison results calculated from Levkovskii's as well as Wen Den Lu et al.'s[13] formulas.

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Table 1. Target nuclei groupea by neutron excess–number and values of the prameters α, β and C best fitted for each grouped data values of 14.1, 14.5 and 14.9 MeV for the (n,p), (n,α) and (n,2n) reactions.

(a) (n,p) Reaction

N-Z	Neutron Energy(MeV)	α (MeV^{-1})	β	C
0	14.1	0.148	3.40	
	14.5	0.162	3.59	
	14.9	0.341	4.15	
1	14.1	-0.185	5.01	90
	14.5	-0.134	5.05	90
	14.9	-0.313	5.42	110
3	14.1	-0.554	16.5	220
	14.5	-0.575	15.9	210
	14.9	-0.631	15.7	210
5.7	14.1	0.0244	5.87	50
	14.5	0.0866	5.97	50
	14.9	0.0172	5.89	50
2.4. 6.8	14.1	0.150	4.44	30
	14.5	0.108	4.24	30
	14.9	0.0999	4.26	30
9.11. 13.15	14.1	-0.664	10.9	90
	14.5	-0.508	7.18	60
	14.9	-0.417	4.76	40
10~54 (even)	14.1	-0.0668	5.21	40
	14.5	0.0456	3.77	30
	14.9	0.0298	5.28	40
17~53 (odd)	14.1	0.194	5.38	40
	14.5	0.0691	5.49	40
	14.9	-0.390	5.10	40

(b) (n,α) Reaction

N-Z	Neutron Energy(MeV)	α (MeV^{-1})	β	C
0.1.2	14.1	-0.134	3.00	20
	14.5	-0.0527	2.67	10
	14.9	-0.0416	2.76	10
3.5. 7.9	14.1	0.0163	4.18	40
	14.5	0.0252	4.16	40
	14.9	0.0217	4.18	40
4.6. 8.10	14.1	-0.108	5.12	50
	14.5	-0.0577	2.09	20
	14.9	-0.0340	3.11	30
11~53 (odd)	14.1	0.119	3.16	40
	14.5	0.108	3.77	40
	14.9	-0.00825	4.23	40
12~20 (even)	14.1	0.0815	5.31	50
	14.5	-0.0379	8.43	70
	14.9	-0.393	9.49	70
22~54 (even)	14.1	-0.161	13.6	80
	14.5	0.0815	4.62	40
	14.9	0.179	6.05	50

(c) (n,2n) Reaction

N-Z	Neutron Energy(MeV)	α (MeV^{-1})	β	C
0.2. 4.6	14.1	0.785	8.24	-50
	14.5	0.388	3.67	-50
	14.9	0.292	2.73	-50
1.3.5	14.1	0.309	1.26	-70
	14.5	0.140	-0.397	-70
	14.9	0.271	1.76	-60
8.10. 12.14	14.1	0.0590	3.21	-10
	14.5	-0.0314	2.32	-10
	14.9	-0.0169	2.59	-10
7.9. 11.13	14.1	0.211	6.16	0
	14.5	0.180	5.90	0
	14.9	0.139	5.53	0
15~53 (odd)	14.1	-1.11	-10.9	-30
	14.5	-0.973	-7.89	-20
	14.9	-0.930	-7.40	-20
16~38 (even)	14.1	0.163	-7.20	10
	14.5	0.0574	4.61	0
	14.9	0.0368	4.44	0
40~54 (even)	14.1	-1.09	-14.8	-50
	14.5	-1.38	-21.2	-70
	14.9	-1.44	-21.7	-70

Table 2. Cross section values calculated from the new formula using the best fitted parameter values and those from other formulas for the (n,p), (n,α) and (n,2n) reactions.

(a) (n,p) Reaction

				14.1 MeV		14.5 MeV		14.9 MeV		Levkovskii[3]
Z	Target	(N-Z)	Q-value	(Exp.)	Cal.	(Exp.)	Cal.	(Exp.)	Cal.	Cal.
6	C - 12	0	-12.59	—	—	—	—	1.93	4.55	489
7	N - 15	1	-8.99	—	—	—	—	16	15.0	60.2
8	O - 16	0	-9.64	41	45.6	39	48.4	34	15.1	560
9	F - 19	1	-4.04	20	19.8	19	16.7	18	17.5	107
11	Na- 23	1	-3.60	43	47.3	34	40.8	43	47.4	159
12	Mg- 24	0	-4.73	110[10]	124	190	140	180	105	682
12	Mg- 25	1	-3.05	63	61.7	44	54.8	40	61.8	186
12	Mg- 26	2	-7.92	27	22.5	—	—	50	27.9	56.1
13	Al- 27	1	-1.83	78	67.7	75	63.9	75	61.5	213
14	Si- 28	0	-3.86	246[10]	156	230	179	210	157	736
14	Si- 29	1	-2.90	—	—	120	97.3	147	119	240
15	P - 31	1	-0.71	88	92.8	82[6]	92.7	83	80.3	267
16	S - 32	0	-0.93	225	262	225	315	212	466	788
16	S - 34	2	-4.32	78	79.5	—	—	73	82.3	117
17	Cl- 35	1	0.62	107	110	120	117	—	—	321
17	Cl- 37	3	-4.07	25	28.4	33	36.9	41	39.9	58.4
18	Ar- 38	2	-4.13	—	—	75	104	110	109	151
18	Ar- 40	4	-6.72	—	—	15.7	19.6	20	21.0	32.6
19	K - 41	3	-1.71	50	46.9	—	—	48	50.8	80.0
20	Ca- 40	0	-0.53	298[10]	323	—	—	—	—	883
20	Ca- 42	2	-2.73	175	162	182	150	180	155	188
20	Ca- 43	3	-1.04	97	70.6	110	76.7	—	—	91.7
20	Ca- 44	4	-4.88	35	33.1	46[8]	33.4	36	35.3	46.2
21	Se- 45	3	0.53	55	60.4	56	61.4	53	51.5	104
22	Ti- 46	2	-1.58	290	232	166[8]	204	270	210	226

(b) (n,α) Reaction

				14.1 MeV		14.5 MeV		14.9 MeV		Levkovskii[3]
Z	Target	(N-Z)	Q-value	(Exp.)	Cal.	(Exp.)	Cal.	(Exp.)	Cal.	Cal.
4	Be- 9	1	-0.60	10	10.2	—	—	—	—	4.38
5	B - 11	1	-6.63	30	38.9	—	—	31	41.6	9.36
11	Na- 23	1	-3.87	150	114	—	—	150	97.5	63.6
12	Mg- 26	2	-5.43	84	77.7	77	78.3	72	80.8	22.4
13	Al- 27	1	-3.13	120.5	130	116	106	111	112	85.2
14	Si- 30	2	-4.20	—	—	70	89.5	—	—	33.8
15	P - 31	1	-1.94	119	134	118	115	115	123	107
16	S - 34	2	-1.33	126	77.3	138	90.5	163	97.6	46.7
17	Cl- 35	1	0.94	100	107	117	111	122	123	128
18	Ar- 40	4	-2.49	13	17.3	10	14.7	10	14.2	13.0
19	K - 39	1	1.36	84	115	—	—	—	—	150
19	K - 41	3	-0.11	46	41.7	39	40.7	31	41.4	32.0
20	Ca- 44	4	-2.74	35	29.8	29[5]	19.1	35	20.1	18.5
21	Se- 45	3	-0.40	55	57.3	56	55.8	54	56.7	41.6
22	Ti- 48	4	-2.03	39	42.7	—	—	23	26.1	24.8
22	Ti- 50	6	-3.44	9.4	8.16	9.5	12.1	10	9.36	7.54
23	V - 51	5	-2.06	15	17.3	17	16.6	19	17.0	15.8
24	Cr- 54	6	-1.55	—	—	—	—	9.0[9]	12.1	10.6
25	Mn- 55	5	-0.63	32	24.7	25[5]	24.0	32	24.5	20.8
26	Fe- 54	2	0.84	100	122	98	137	96	151	122
26	Fe- 58	6	-1.39	—	—	—	—	21.5	15.8	14.1
27	Co- 59	5	0.32	30	33.7	30	33.0	29	33.5	26.4
28	Ni- 58	2	2.89	123	102	—	—	—	—	137
28	Ni- 62	6	-0.44	22	21.8	—	—	17.2	19.6	18.2
28	Ni- 64	8	-2.43	—	—	—	—	5.2	9.17	7.31
29	Cu- 63	5	1.72	39.1	44.6	35.7	44.3	55.7	44.7	32.7
29	Cu- 65	7	-0.09	—	—	14[7]	13.9	11.9	14.1	13.0

(c) (n,2n) Reaction

				14.1 MeV		14.1 MeV		14.9 MeV		Lu and Fink[13]
Z	Target	(N-Z)	Q-value	(Exp.)	Cal.	(Exp.)	Cal.	(Exp.)	Cal.	Cal.
9	F - 19	1	-10.43	47	39.9	55	44.4	60	57.7	139
11	Na- 23	1	-12.42	28	12.9	44	20.1	38.5[11]	22.1	89.3
15	P- 31	1	-12.31	5.1	7.44	10.9	11.4	10	14.1	5.47
19	K - 39	1	-13.09	2.5	4.29	3.5	7.47	5.9	8.97	-64.2
20	Ca- 48	8	-9.95	900	962	920	968	1070	1103	916.6
21	Sc- 45	3	-11.32	267	144	331	186	354	187	337
22	Ti- 46	2	-13.20	13.5	13.5	30	26.5	50	36.5	127
24	Cr- 50	2	-12.94	10	14.7	26.4	26.1	28	35.0	95.0
24	Cr- 52	4	-12.04	278	194	—	—	358	296	451
25	Mn- 55	5	-10.22	855	1261	890	1352	850	1232	575
26	Fe- 54	2	-13.38	10.5	9.46	15.5	19.9	22	27.9	64.6
26	Fe- 56	4	-11.20	440	300	—	—	—	—	421
27	Co- 59	5	-10.46	655	797	720	890	735	836	548
28	Ni- 58	2	-12.20	24.5	22.1	31	29.1	34.9	36.4	35.6
29	Cu- 63	5	-10.85	480	507	522	604	585	569	521
29	Cu- 65	7	-9.90	913	943	956	991	975	1027	754
30	Zn- 64	4	-11.86	119	125	165	144	204	174	364
30	Zn- 66	6	-11.05	620	996	650	832	740	932	633
30	Zn- 70	10	-9.22	—	—	—	—	1307	1104	1005
31	Ga- 69	7	-10.31	850	900	957	957	1070	1010	732
32	Ge- 70	6	-11.53	508	548	610	554	700	650	608
32	Ge- 76	12	-9.44	1157	1234	1210	1186	1232	1360	1129

IV. Kyoto University

Research Reactor Institute

IV-1 Identification of a New Isotope ^{156}Pm

K. Okano, Y. Kawase and Y. Funakoshi

A paper on this subject was published in J. Phys. Soc. Jpn. 55 (1986) 715—718 with the following abstract:

A new nuclide ^{156}Pm has been identified among the fission products of ^{235}U using the on-line isotope separator(KUR-ISOL) for mass separation and identification. The atomic number has been identified by the energies of X-rays and γ-rays emitted. The half-life of 29 $\pm$2 s obtained for ^{156}Pm is consistent with the prediction based on the gross theory of β-decay of Takahashi et al.

IV-2

Identification of a New Isotope ^{155}Nd

K. Okano, Y. Kawase and K. Aoki

A paper on this subject is in press as Annu. Rep. Res. Reactor Inst. Kyoto Univ. Vol. 19.(1986). The contents may be summarized as follows:

The heaviest isotope of neodymium, ^{155}Nd, has been identified among the fission products of ^{235}U using the He-jet fed on-line isotope separator KUR-ISOL.[1,2] The Z-identification has been performed by the method of ion source chemistry using a high-temperature thermal ion source, as well as by measuring the energies of X-rays. Pm K X-rays and four γ-rays have been found to be associated with the decay of ^{155}Nd. The energies, relative intensities and half-lives of γ-rays observed are shown in Table 1. The decay curves of three γ-rays and the Pm K_α X-ray (observed as 38.4 ± 0.4 keV) are shown in Fig. 1. The half-life of the Pm K_α X-ray was determined as 10.0 ± 2.0 s. The half-life of ^{155}Nd obtained by averaging these half-lives, 9.5 ± 0.7 s, is shorter than theoretically predicted values.[3-5]

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Table 1. Energies, relative intensities and half-lives of γ-rays observed in the decay of ^{155}Nd.

Energy (keV)	Relative intensity	Half-life (s)
67.5 ± 0.3	29 ± 5	11.5 ± 4.5
180.7 ± 0.2	100 ± 6	9.2 ± 0.9
418.9 ± 0.3	82 ± 12	9.6 ± 1.3
955.1 ± 0.3	48 ± 16	10.0 ± 1.5

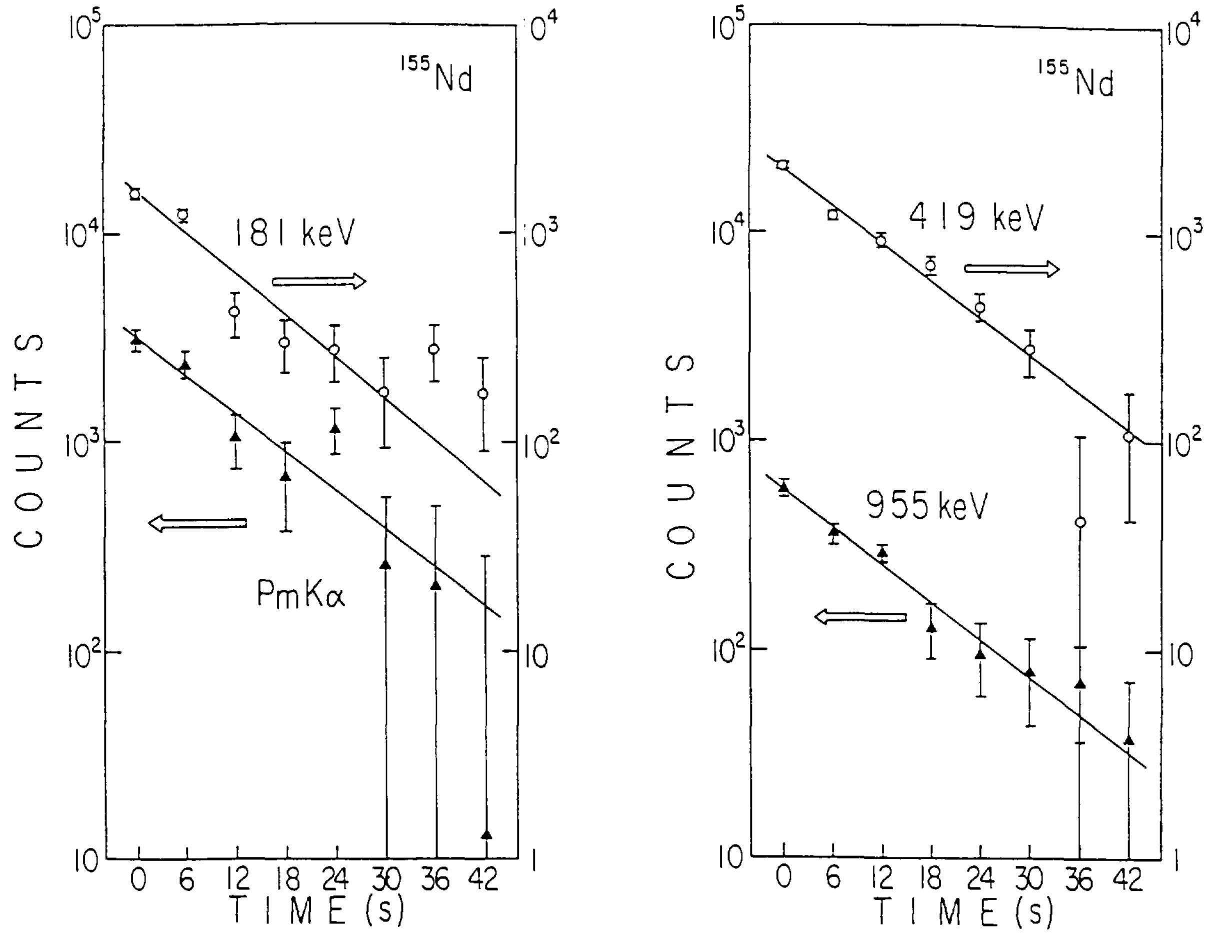

Fig. 1. Decay curves of Pm K_α X-ray and three γ-ray lines assigned to originate from the decay of ^{155}Nd.

IV-3

Gamma-ray Emission Probabilities at Mass 95 and Log f_1t Values of Y Isotopes

K. Okano, Y. Funakoshi and Y. Kawase

A paper of this subject was published in J. Phys. G.: Nucl. Phys. 12 (1986) 737 — 744 with the following abstract:

Using the ^{95}Rb activity mass-separated by KUR-ISOL, the absolute emission probabilities of the γ-rays of ^{95}Rb at 352.0 keV, ^{95}Sr at 685.6 keV and ^{95}Y at 954.2 keV have been determined as 49.2 $\pm$ 3.1, 21.8 $\pm$ 1.8 and 15.9 $\pm$ 0.7 per 100 decays, respectively, by an affiliation method. The log f_1t value for the first-forbidden unique β^- decay of ^{95}Y to the ground state of ^{95}Zr has been determined as 8.62 $\pm$ 0.04. The systematics of similar transitions in Y isotopes are compared with theoretical predictions.

Half-life Measurements of ^{153}Nd and ^{154}Nd Mass-separated by KUR-ISOL

K. Okano, Y. Kawase and K. Aoki

The half-life of ^{153}Nd has previously been reported to be 32 $\pm$ 4 s by Pinston et al.[1] No other report has since appeared on the measurement of its half-life. The half-life of ^{154}Nd has been believed to be 40 s for a long time,[2,3] but recently Karlewski et al. reported a revised value of 26 $\pm$ 2 s as the half-life of a newly identified ^{154}Nd.[4] We have measured the half-lives of ^{153}Nd and ^{154}Nd using a 142 cc Ge(Li) detector and a He-jet fed on-line isotope separator KUR-ISOL with a high-temperature thermal ion source. The decay curves of the most prominent γ-rays, 418 keV γ-ray for ^{153}Nd and 152 keV γ-ray for ^{154}Nd, have been analyzed. The half-life of ^{153}Nd has been determined as 29.5 $\pm$ 0.8 s and that of ^{154}Nd as 26.9 $\pm$ 1.0 s. Figure 1 shows the decay curve of the 418 keV γ-ray of ^{153}Nd.

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2) K. Buchtela, Atomkernenergie 22 (1974) 268.
3) Chart of the Nuclides, ed. Y. Yoshizawa, T. Horiguchi and M. Yamada, Japan Atomic Energy Researsh Institute, Tokai, 1980, 1984.
4) T. Karlewski, N. Hildebrand, G. Herrmann, N. Kaffrell, N. Trautmann and M. Brügger, Z. Phys. A 322 (1985) 177.

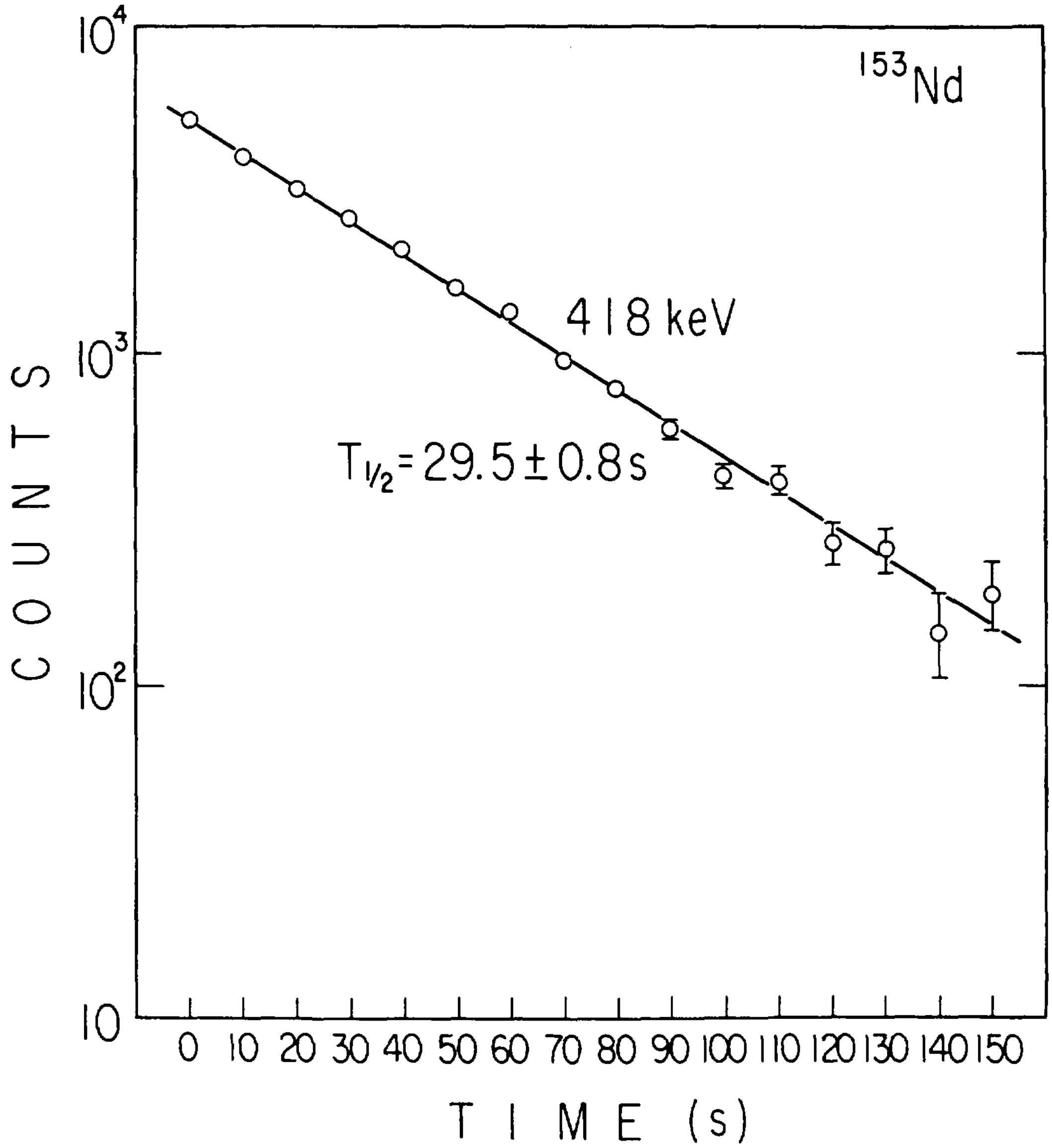

Fig. 1. Decay curve of the 418 keV γ-ray following the decay of ^{153}Nd.

IV-5

Measurement and Analysis of Neutron Spectra In Structural Materials for Reactors

Itsuro Kimura, Shu A. Hayashi, Katsuhei Kobayashi,
Shuji Yamamoto, Hiroshi Nishihara*, Satoshi Kanazawa*,
Takamasa Mori** and Masayuki Nakagawa**

In order to assess evaluated nuclear data of main structural materials for fission and fusion reactors, measurement and analysis of energy spectra of neutrons in sample piles or scattered by sample slabs have been continuously carried out.

(1) The final result of the neutron spectrum in the copper pile was published recently(1).

(2) The neutron spectrum in a spherical pile of silicon was measured and analyzed. Silicon granule, whose purity was 99.9%, was packed with the apparent density of 1.3 g/cm^3 into a steel vessel. The inner diameter and thickness of the vessel were 60 cm and 4.5 mm, respectively. A photoneutron target was placed at the center of the pile and the neutrons at $r = 15$ cm (from the center) and $\mu = 0$ (90^0 to the radial direction) were extracted from the reentrant hole of the pile to the direction of the neutron detector. We measured the energy spectrum of the neutrons by the Linac-TOF method. The details of the experimental

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arrangement can be seen elsewhere(2). It can be seen that both predictions with the ENDF/B-IV and the JENDL-2 are close to the measured value from several tens of keV to a few MeV, although the statistical error below 144 keV is rather large. The preliminary result was presented before(3) and its final one will be submitted soon.

(3) The results of the neutron spectra in the iron, nickel and chromium piles were presented at the Consultants' Meeting on Evaluated Neutron Cross Section Data for Structural Materials(4), and then they have been refined and will be submitted to J. Nucl. Sci. Technol.(5).

The experimental results were compared with the theoretical values derived from either of the cross-section evaluations JENDL-2 and ENDF/B-IV by one-dimensional transport calculation. The findings are as follows: (1) For both of the evaluations, the resonance parameters of Fe and Ni in the energy range below 100 keV should be revised. (2) The inelastic scattering cross-section data of Fe should be added to the ENDF/B-IV edition below 840 keV. (3) Reevaluation of the total cross-section of Cr is necessary in the energy range from 4 to 8 keV where there are a series of big resonances. In addition to the TOF experiments, neutron transmission measurement was carried out with chromium samples. The result indicates that the total cross-section should be large by 30 to 40 % than both of the JENDL-2 and the ENDF/B-IV evaluation.

Reference:

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(2) T.Mori,et al.,J.Nucl.Sci.Technol.Vol.20, No.12, 991(1983).

(3) I. Kimura, et al., Int. Conf. on Nuclear Data for Basic and Applied Science, Santa Fe (1985) to be published in radiation effect.

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IV-6

Measurement of Self-Shielding Factor of Neutron Capture Cross Section for ^{181}Ta in Unresolved Resonance Region

I.Kimura, Y.Fujita, K.Kobayashi, S.Yamamoto,
H.Oigawa* and S.Kanazawa*

The self-shielding factor of neutron capture cross section has been measured for ^{181}Ta in the unresolved resonance region by a neutron time-of-flight spectrometer at the electron-linac facility of Kyoto University Research Reactor Institute. The purpose of the measurement is to access the average resonance parameters used in the calculation of the self-shielding factor.

The factor obtained by the measurement is the type of Bondarenko[1] and is defined by

$$f_{n\gamma}(\sigma_0) = \frac{\left\langle \dfrac{\sigma_{n\gamma}}{\sigma_t + \sigma_0} \right\rangle}{\left\langle \dfrac{1}{\sigma_t + \sigma_0} \right\rangle} \cdot \frac{1}{\langle \sigma_{n\gamma} \rangle}$$

where σ_t , $\sigma_{n\gamma}$, σ_0 are total, capture and dilution cross sections respectively, and the bracket means the average in an energy group.

The factor is deduced for an arbitrary σ_0 from a set of experimental data of the standard transmission and self-indication measurements.[2]

The data for ^{181}Ta have been taken and now in processing. The measurement will be extended to ^{238}U in the near future.

1) I.I.Bondarenko edit.; "Group Constants for Nuclear Reactor Calculations" Consaltants Bureau, New York (1964).

2) A.Arnaud et al.; Proc. of "Nuclear Cross Sections and Technology", NBS Special Publication 425 , (1975).

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IV-7

Resonance Parameters and Resonance Integral of ^{232}Th

Katsuhei Kobayashi and Yoshiaki Fujita

Nuclear data of ^{232}Th, especially the resonance parameters are of great importance for the safe and economical design of nuclear reactors utilizing thorium-based fuels. The first four s-wave resonances of ^{232}Th give about 70 % of the resonance integral for the $^{232}Th(n,\gamma)$ reaction.

Transmission measurements of metallic Th-samples were made by the time-of-flight(TOF) method using the 46 MeV electron linear accelerator(linac) at the Research Reactor Institute, Kyoto University(KURRI). The experimental method is similar to that in the previous work[1]. Neutron transmission rates from thorium samples(1.5, 2.9, 9.1, 18.4, 27.7 x 10^{-3} atoms/barn) were measured, and the resonance parameters were deduced from the area analysis by Hughes' method[2] and the shape analysis by the SIOB code[3]. The results obtained by the area analysis were also employed as the initial values used in the SIOB calculation.

The results obtained are shown in Table 1, where the evaluated resonance parameters in JENDL-2, ENDF/B-IV and ENDF/B-V are also given. The present results agree well with the recent works by Olsen[4] and Chrien[5], except Γ_γ at 69.2 eV for the area analysis.

In order to make an integral assessment of the resonance parameters of ^{232}Th, the resonance integral has been calculated by using the present data from the area analysis and the shape analysis. To do so, we have exchanged the first-four ^{232}Th s-wave resonance parameters in JENDL-2 and in ENDF/B-IV for those in the present measurements. The codes RESENDD[6] and INTERN[7] were used for these calculations. The calculated resonance integrals have been compared with the measured value which was obtained in the 1/E standard neutron field of the Research Reactor, UTR-Kinki[8].

These results and the resonance integrals from JENDL-2 and ENDF/B-IV are summarized in Table 2. The calculations with the present parameters are close to the measurement. However, the results with JENDL-2 resonance parameters show lower values, in general. This fact may come from the smaller values of ^{232}Th parameters in JENDL-2, especially at lower energies, as seen in Table 1.

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4) D. K. Olsen, et al., ORNL/TM-7661 (1981).

5) R. E. Chrien, et al., Nucl. Sci. Eng., 72, 202 (1979).

6) T. Nakagawa, JAERI-M 84-192 (1984).

7) T. Nakagawa, Private communication (1986).

8) K. Kobayashi, et al., NEANDC(J)-116/U, p.56 (1985).

Table 1 Widths (meV) for the first-four ^{232}Th s-wave resonances

Measurement or evaluation	21.8 eV Γ_n	21.8 eV Γ_γ	23.5 eV Γ_n	23.5 eV Γ_γ	59.5 eV Γ_n	59.5 eV Γ_γ	69.2 eV Γ_n	69.2 eV Γ_γ
JENDL-2	1.913	20.00	3.243	25.00	3.933	25.00	44.00	25.00
ENDF/B-IV	2.00	25.9	3.74	25.9	4.00	25.9	42.0	25.9
ENDF/B-V	2.02	23.0	3.88	25.0	4.04	23.2	44.0	21.9
Area analysis	2.07±.14	24.4±2.1	4.00±.29	25.0±2.2	3.86±.27	26.0±2.9	41.7±6.6	18.3±3.5
Shape analysis	2.09±.03	25.2±0.7	3.88±.18	26.1±1.4	3.83±.21	25.0±2.1	42.6±1.3	22.9±1.3

Table 2 Resonance integral of ^{232}Th

Resonance integral (b)	Reference
86.2 ± 3.1	Authors(1985) 1/E standard field
79.93	JENDL-2
85.60	ENDF/B-IV
JENDL-2: base	Exchanged
84.36	(1) Shape analysis data
82.87	(2) Area analysis data
84.03	(3) ENDF/B-IV
83.47	(4) ENDF/B-V
ENDF/B-IV: base	Exchanged
85.91	(5) Shape analysis data
84.42	(6) Area analysis data
81.49	(7) JENDL-2
85.03	(8) ENDF/B-V

IV-8

Cf-252 Fission Neutron Spectrum as an Integral Field[1)]

Itsuro Kimura, Katsuhei Kobayashi and Osamu Horibe*

By making use of the average cross section data(Mannhart's evaluation)[2)] for twenty five reactions measured with the ^{252}Cf spontaneous fission neutron source, its spectral shape was unfolded by NEUPAC. As a whole, the result agrees with the predicted ones by Madland and by Märten and with the measured one by Pönitz and Tamura, but obviously differs from the Maxwellian spectrum with its average energy E_{av}=2.13 MeV, although the unfolded spectrum shows a little oscillatory structure.

The ^{252}Cf spectrum averaged cross sections for the above reactions were calculated by making use of energy dependent cross section data from ENDF/B-V (dosimetry file), JENDL-2 and internal library of NEUPAC and of six expressions of its spectral shape; (1) Maxwellian(E_{av}=2.13 MeV), (2) Madland's prediction[3)], (3) Märten's predictions(CEM and GMNM)[4)], (4) Gerasimenko's prediction[5)], and (5) the spectrum unfolded by NEUPAC. The result is tabulated in Table 1. From the comparison of the calculated values with the measured ones, it can be seen that the ranking of the agreement is as follows: (1) Unfolded by NEUPAC, (2) Märten's(CEM), (3) Märten's(GMNM), (4) Gerasimenko's, (5) Madland's and (6) Maxwellian.

The average cross section data for the above reactions were tried to be plotted by the Horibe's empirical rule[6)]. Al-

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though the number of data is rather restricted, we can see good systematics for the (n,p), (n,α) and (n,2n) reactions similar to the case of the average cross sections to the ^{235}U fission neutron spectrum.

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5) B. F. Gerasimenko and V. A. Rubchenya: ibid., p.280.
6) O. Horibe: Ann. nucl. Energy, 10, 359 (1983).

Table 1 Comparison of the C/E ratios of the Cf-252 spectrum-averaged cross sections

Reaction	Ratio of calculated value to experimental value (Mannhart's evaluation)										
	Maxwellian E_{av}=2.13 MeV		Madland's prediction		Märten's (GMNM) prediction		Märten's (CEM) prediction		Gerasimenko's prediction		Unfolded by NEUPAC
	ENDF/B-V	JENDL-2	ENDF/B-V	JENDL-2	ENDF/B-V	JENDL-2	ENDF/B-V	JENDL-2	ENDF/B-V	JENDL-2	Internal library
$^{27}Al(n,p)^{27}Mg$	1.080	0.953	1.018	0.900	0.996	0.880	1.010	0.892	1.023	0.903	1.031
$^{27}Al(n,\alpha)^{24}Na$	1.139	1.151	0.965	0.979	0.982	0.994	0.976	0.989	1.032	1.045	1.010
$^{46}Ti(n,p)^{46}Sc$	0.980		0.925		0.904		0.917		0.927		0.935
$^{47}Ti(n,p)^{47}Sc$	1.256		1.255		1.225		1.241		1.244		1.235
$^{48}Ti(n,p)^{48}Sc$	1.049		0.894		0.909		0.905		0.946		0.937
$^{55}Mn(n,2n)^{54}Mn$	1.366	1.393	0.950	0.969	1.117	1.139	1.079	1.100	0.993	1.101	1.057
$^{54}Fe(n,p)^{54}Mn$	1.028	0.956	1.021	0.941	0.994	0.916	1.010	0.931	1.009	0.931	1.001
$^{56}Fe(n,p)^{56}Mn$	1.029	1.069	0.910	0.945	0.908	0.943	0.911	0.945	0.945	0.982	0.946
$^{58}Ni(n,p)^{58}Co$	0.986	0.958	0.981	0.954	0.956	0.929	0.970	0.943	0.970	0.943	0.955
$^{58}Ni(n,2n)^{57}Ni$	1.112	1.095	0.688	0.677	0.903	0.890	0.868	0.855	0.681	0.669	0.883
$^{59}Co(n,\alpha)^{56}Mn$	1.074	1.147	0.921	0.988	0.933	0.999	0.930	0.997	0.974	1.042	0.951
$^{59}Co(n,2n)^{58}Co$	1.262	1.167	0.869	0.807	1.030	0.954	0.994	0.920	0.905	0.842	0.987
$^{63}Cu(n,\alpha)^{60}Co$	1.185	1.131	1.050	1.008	1.048	1.004	1.051	1.008	1.093	1.045	1.076
$^{65}Cu(n,2n)^{64}Cu$	1.224	1.224	0.868	0.864	1.005	1.004	0.972	0.970	0.913	0.909	0.951
$^{115}In(n,n')^{115m}In$	0.921		0.941		0.924		0.929		0.934		0.928
$^{127}I(n,2n)^{126}I$	1.332		1.001		1.105		1.075		1.076		1.077
$^{19}F(n,2n)^{18}F$		1.619		1.074		1.319		1.271		1.101	1.001
$^{24}Mg(n,p)^{24}Na$											1.053
$^{63}Cu(n,\gamma)^{64}Cu$											0.943
$^{63}Cu(n,2n)^{62}Cu$		1.460		0.952		1.186		1.141		0.969	1.065
$^{64}Zn(n,p)^{64}Cu$											0.952
$^{90}Zr(n,2n)^{89}Zr$											0.965
$^{115}In(n,\gamma)^{116m}In$											0.988
$^{197}Au(n,\gamma)^{198}Au$											1.012
$^{197}Au(n,2n)^{196}Au$											1.032
$\frac{\sum_i^N \lvert 1.0-(C/E)_i \rvert}{N}$	0.1405	0.1992	0.0895	0.0851	0.0701	0.0882	0.0683	0.0827	0.0806	0.0845	0.0532

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V-A-1

SHELL AND ODD-EVEN EFFECTS IN PRE-EQUILIBRIUM (p,p') SPECTRA FOR NUCLEI AROUND NEUTRON NUMBER 50

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In order to clarify the shell effect and the odd-even effect in the pre-equilibrium process on the (p,p') reaction, we have undertaken to measure systematically and accurately the double differential cross sections of the (p,p') reaction which is analogous one with the (n,n') reaction. By the analogy with the (p,p') reaction, we expect to clarify the shell effect and the odd-even effect on the 14 MeV (n,n') reaction.

The experiment was performed using 18 MeV proton beams from the tandem Van de Graaff accelerator at Kyushu University. All the targets were self-supporting metallic foils whose thickness were about 0.5 mg/cm^2. The emitted protons were detected using a silicon ΔE-E counter telescope (ΔE: 75 μm, E: 2500 μm), and the solid angle subtended by the telescope was 0.89 msr. Signals of proton were separated from those of other charged particles with a particle identification circuit. The proton

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spectra were measured at the angles ranging from 40° to 160° in steps of 10°.

The angle-integrated spectra of protons emitted from (p,p') reactions with ^{90}Zr, ^{92,94,96,98,100}Mo, ^{93}Nb, ^{106}Pd and Ag are shown in fig.1. The spectra for outgoing energies larger than 14 MeV were excluded from analyses by the pre-equilibrium model because they show pronounced structures corresponding to collective excitations, such as the lowest 2^+ and 3^- states. The spectra for ^{90}Zr, ^{92}Mo and ^{94}Mo indicate relatively strong evaporation peaks at low outgoing proton energy; particulary, contributions of protons from (p,np) and/or (p,2p) processes are large in the outgoing energy range below 6 MeV for ^{90}Zr and below 5 MeV for ^{92}Mo.

For nuclei apart from the magic nuclei, the spectra exhibit trends to be more flat and smooth with increasing the neutron number, and have the similar shape and almost the same magnitude within several % in the continuum region of 10-14 MeV. The cross sections for ^{92}Mo and ^{94}Mo are larger than those for the other targets at the outgoing energy between 10 and 14 MeV, which can be explained by a large equilibrium component.

Table 1 indicates comparisons of the cross section (σ_{remain}) integrated over 10-14 MeV that is obtained by subtraction of the calculated equilibrium cross section (σ_{eq}) from the experimental one (σ_{exp}) for each target nucleus. The quantity σ_{eq} for each target was derived from the calculation using separate normalizations for the equilibrium and pre-equilibrium components so as to reproduce each experimental spectrum. The quantity σ_{remain} is considered to be the non-equilibrium component, which mainly consists of the pre-equilibrium component. From comparisons of σ_{remain}, it was found that pre-equilibrium component is nearly constant for all the isotopes of Mo within the experimental errors ($\sim$5 %).

Studies of the (p,p') reaction with near 18 MeV protons on tin isotopes, which are nuclei far apart from the neutron magic number, have been reported in ref. [1,2]. The measured energy spectra for tin isotopes show almost the similar shape and have the nearly equal magnitude in the excitation energy range of 4 to 8 MeV, except that for ^{112}Sn with a large equilibrium component. This tendency is the same as that for Mo isotopes as mentioned above.

These facts indicate that there are no appreciable shell effect on the pre-equilibrium cross section of the (p,p') reaction averaged over the outgoing proton energy spectra corresponding to excitations higher than 4 MeV.

Furthermore, the odd-even effect on the pre-equilibrium process has been investigated by comparing experimental results of the odd-even pairs of ^{93}Nb - ^{94}Mo, and Ag - ^{106}Pd. The proton spectra for ^{93}Nb and ^{94}Mo show the different shape and magnitude in the whole energy region. This seems to be due to the difference of contributions from the equilibrium process. Subtraction of the equilibrium component from the experimental data brings almost the same cross section for both the spectra as shown in table 1. The proton spectra for Ag and ^{106}Pd also exhibit the same magnitude and shape above 10 MeV in fig.1. No appreciable odd-even effect, therefore, was found in the pre-equilibrium (p,p') spectra corresponding to excitations higher than 4 MeV.

In conclusion, it was found that the measured (p,p') spectra exhibit no appreciable shell and odd-even effects in the excitation energy range larger than 4 MeV, where the pre-equilibrium process is dominant. The features of the pre-equilibrium process that was found in the (p,p') reaction would be expected to be observed also in the (n,n') reactions.

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Table 1. Cross sections integrated over 10-14 MeV.
The σ_{exp} stands for the experimental cross section, and the σ_{eq} for the calculated equilibrium cross section. The σ_{remain} is equal to $\sigma_{exp} - \sigma_{eq}$.

nuclei	σ_{exp} (mb)	σ_{eq} (mb)	σ_{remain} (mb)
^{90}Zr	48.44	2.24	46.20
^{93}Nb	50.71	2.99	47.72
^{92}Mo	66.20	14.86	51.34
^{94}Mo	56.25	5.45	50.80
^{96}Mo	50.58	1.32	49.26
^{98}Mo	50.44	0.67	49.77
^{100}Mo	49.55	0.45	49.10
^{106}Pd	46.33	0.53	45.80
Ag	47.63	1.33	46.30

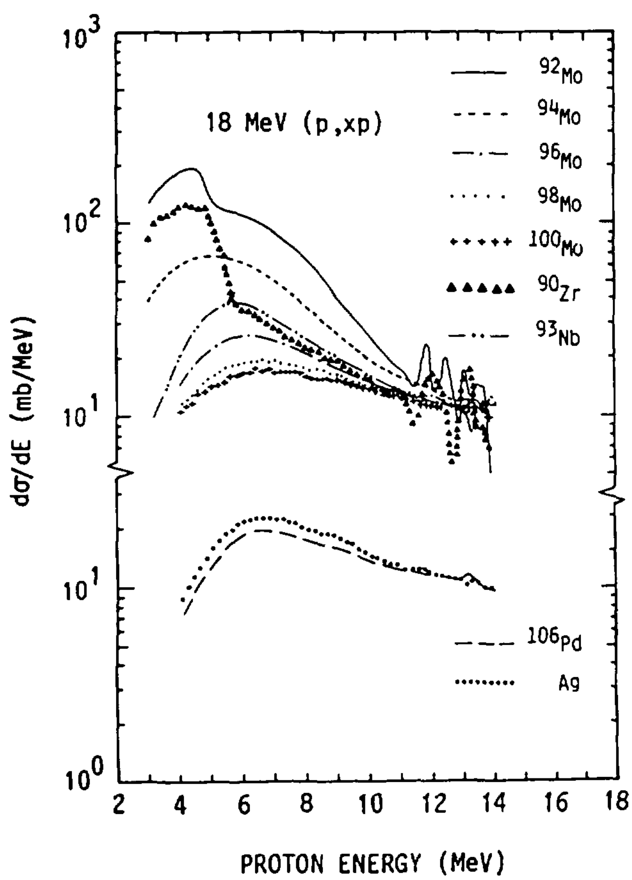

Fig.1 Angle-integrated energy spectra of protons emitted from the (p,p') reaction on ^{90}Zr, ^{92,94,96,98,100}Mo, ^{93}Nb, ^{106}Pd, Ag at the bombarding energy of 18 MeV.

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V-B-1 Sensitivity analysis of Ni,Co(n,x) cross sections and estimation of optical model parameter

T. Yugawa, Y. Uenohara and Y. Kanda

Level density parameters and energy shifts have been estimated by using Bayesian method in the previous works. In Hauser-Feshbach model calculations, the optical model parameters are important as well as level density parameters.

The sensitivity coefficients of cross sections to optical model parameters are required for the estimation. The sensitivity coefficients were calculated by using the optical model calculation code ELIESE-3[1] and Hauser-Feshbach calculation code GNASH[2]. The estimated optical parameters are the depths and radii of both real and imaginary potential. For neutron and proton, the imaginary potential is derivative Wood-Saxon type and for α-particle Wood-Saxon type. It has been found that the (n,x) cross sections were as sensitive to the optical parameters as to level density parameters.

1) S.Igarashi, "Program ELIESE-3" JAERI-1224 (1972)
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V-B-2 Sensitivity of Tritium Production Rate and Neutron Leakage Spectrum to Lithium Neutron Cross Sections

H. Tsuji, Y. Uenohara, T. Shiba, H. Nakashima, and Y. Kanda

Evaluated neutron cross sections play important role on design of nuclear fusion reactors. The cross sections have been adjusted to reproduce integral experimental data such as tritium production rate. A Bayesian method has been utilized to adjust evaluated cross sections. Sensitivities of integral data to cross sections are basic parameters for the Bayesian method. Leakage spectra from an integral experimental system give us more detail information on the cross sections than tritium production rate dose.

The generalized perturbation theory has been applied to the sensitivity analysis of tritium production rate. The generalized perturbation theory, however, is not efficient to sensitivity analysis of neutron leakage spectra. We have, therefore, tried the sensitivity analysis of neutron spectra and tritium production rate. The sensitivity to Li neutron cross sections has been calculated for a natural Li sphere of 30 cm diameter.

V-B-3 Measurement of Helium Production Cross Section of Aluminium for 14.8 MeV Neutrons

T. Fukahori, Y. Kanda, H. Tobimatsu, Y. Maeda,
K. Yamada, T. Nakamura*, and Y. Ikeda*

The He-production cross section of Al for 14.8 MeV neutrons have been measured by Grimes et al.[1] using the magnetic quadrupole spectrometer and by Kneff et al.[2] with the He accumulation method . However, there is a discrepancy by 20 % between both. The present experiment has been performed to solve the problem.

Aluminium samples were irradiated with Fusion Neutronics Source at JAERI. The number of He atoms produced in the sample by $(n,x\alpha)$ was measured with the He-accumulation method. The result is 141 ± 8 mb referring the evaluated $^{27}Al(n,\alpha)^{24}Na$ cross section which is given to be 112.3 ± 0.7 mb by Vonach[3].

This was published in ref. 4.

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VI-1 Measurement of 14 MeV Neutron Activation Cross-sections of Fusion Reactor Materials

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Measurement of 14 MeV neutron activation cross-sections of fusion reactor materials have been done to provide the basic data for the assessment of damage and activation of materials due to the high flux fast neutron.

Molybdenum sample materials were irradiated with the fast neutron obtained by using the Intense Neutron Source (OKTAVIAN) at Osaka University. Gamma-rays from produced radioactive elements were measured with a Ge(Li) detector to obtain cross-sections. The energy dependence of cross-sections was measured for the energy range of 13.5 MeV to 14.9 MeV. Since the neutron energy depends on the emitted angle, sample foils were placed around the Tritium target at the angles

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between 0 and 150 degrees. Distances between samples and the T-target were 15 cm.

Each sample consists of a 0.1 mm thick molybdenum foil between two 0.1 mm thick niobium foils and a 0.1 mm thick zirconium foil. The neutron energy at the irradiation point was estimated from the yield ratio of the $^{90}Zr(n, 2n)$ and the $^{93}Nb(n, 2n)$ reactions. The neutron flux was estimated from the yield of $^{93}Nb(n, 2n)$ reaction by assuming the cross-section as 464 mb.

Before the gamma-ray intensity measurement, the sum coincidence effects in the gamma-ray spectrum was investigated and the method of correction of this effect was established. Then, gamma-ray measurement at a short source-to-detector distance was made to increase the counting rate.

In Fig. 1, the energy dependence of the cross-sections of the $^{92}Mo(n, n'\alpha)$ reaction is shown, and these cross-sections are small. The measurement of these cross-sections can be done by the use of an intense neutron source such as OKTAVIAN and by applying the correction method for the sum coincidence effects studied in this experiment. In Fig. 2, the cross-sections of $^{98}Mo(n, \alpha)$ reaction are shown together with the evaluated values of JENDL-2 which are shown by a solid line in the figure. Analyses of data of other reactions are in progress.

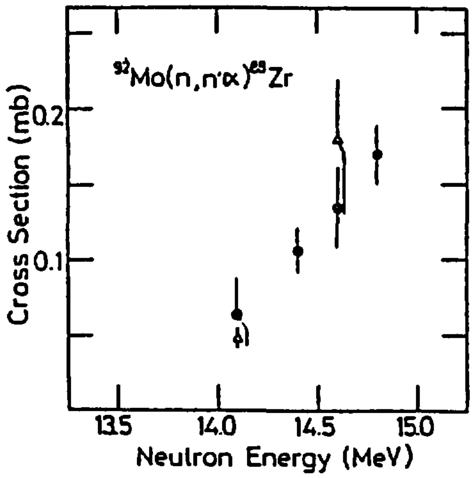

Fig. 1

Activation cross-sections for the ^{92}Mo(n,n'α) reaction.

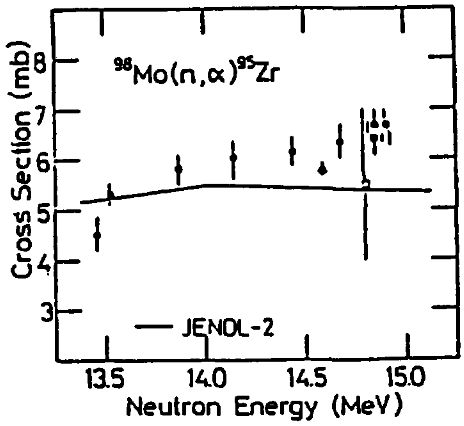

Fig. 2

Activation cross-sections for the ^{98}Mo(n,α) reaction.

IV-2 Half-Lives of Levels in ^{93}Sr and ^{95}Sr

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K. Mio, T. Katoh, J-Z. Ruan*, K. Okano**, Y. Kawase**,

K. Sistemich***, G. Battistuzzi*** and H. Lawin***

A paper on this subject has been published in J. Phys. Soc. Japan vol. 55 no. 4 (1986) pp1102-1107 with an abstract as follows.

The half-life and the multipolarity for the 213 keV transition in ^{93}Sr have been determined to be (4.6 ± 0.3)ns and E2($\leq$ 15 % M1) from β-γ delayed and X-γ coincidences, respectively. The half-life for the 204 keV transition in ^{95}Sr was also determined to be (20.9 ± 0.5)ns. The reduced transition probabilities are deduced as $B(E2) = (261 ^{+17}_{-43})\ e^2fm^4$ for the 213 keV transition and $B(E2) = (71 \pm 2)\ e^2fm^4$ for the 204 keV transition. These B(E2) values are comparable to the Weisskopf estimates and to those for the similar transitions in the neighbouring nuclei.

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Evaluation of Gamma-ray Production Nuclear Data of Ni, Cu, Hf, and U-235

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Gamma-ray Production Nuclear Data are evaluated for Ni (-58 and -60), Cu (-63 and -65), Hf (-174 and -176 through -180), and U-235 in the neutron energy range from 10^{-5} eV to 20 MeV with a preequilibrium and statistical nuclear model code GNASH[1], which was modified to accept a spin selected level density formula and to calculate fission exit channel.

Level density and optical potential parameters are carefully determined so as to well reproduce the particle emission cross sections and spectra. Direct inelastic scattering cross sections[2] were input for the GNASH calculation of Cu isotopes. Fission transmission coefficients for uranium isotopes were determined from the evaluated neutron cross sections and partly from the double humped barrier model[3]. As a consequence, evaluated gamma-ray production nuclear data as well as particle emission spectra agreed well with experimental data[4-6] as is shown in Figs. 1 through 5.

Evaluated gamma-ray production nuclear data were edited for inclusion in the Japanese Evaluated Nuclear Data Library Version-3 with a computer program GAMFIL, which directly reads GNASH results and converts them into ENDF/B format. For this purpose, some subroutines of GNASH were modified.

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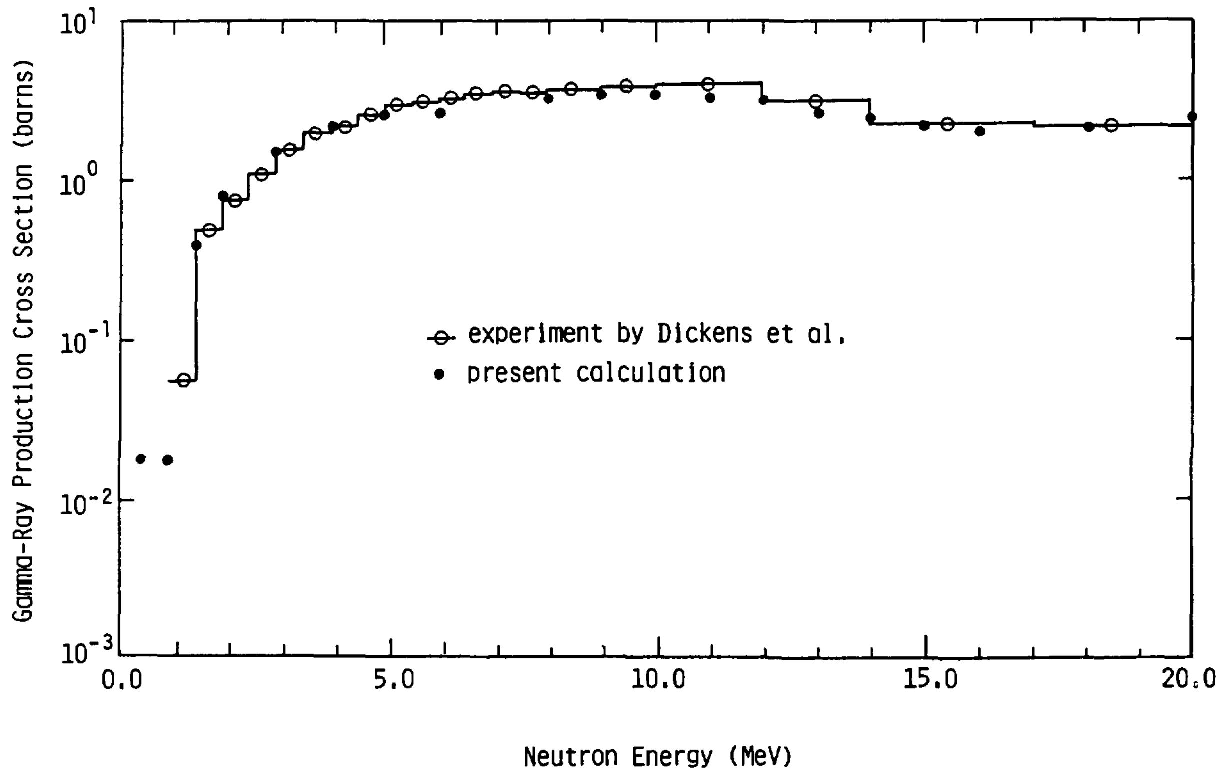

Fig. 1 Total gamma-ray production cross sections for natural nickel

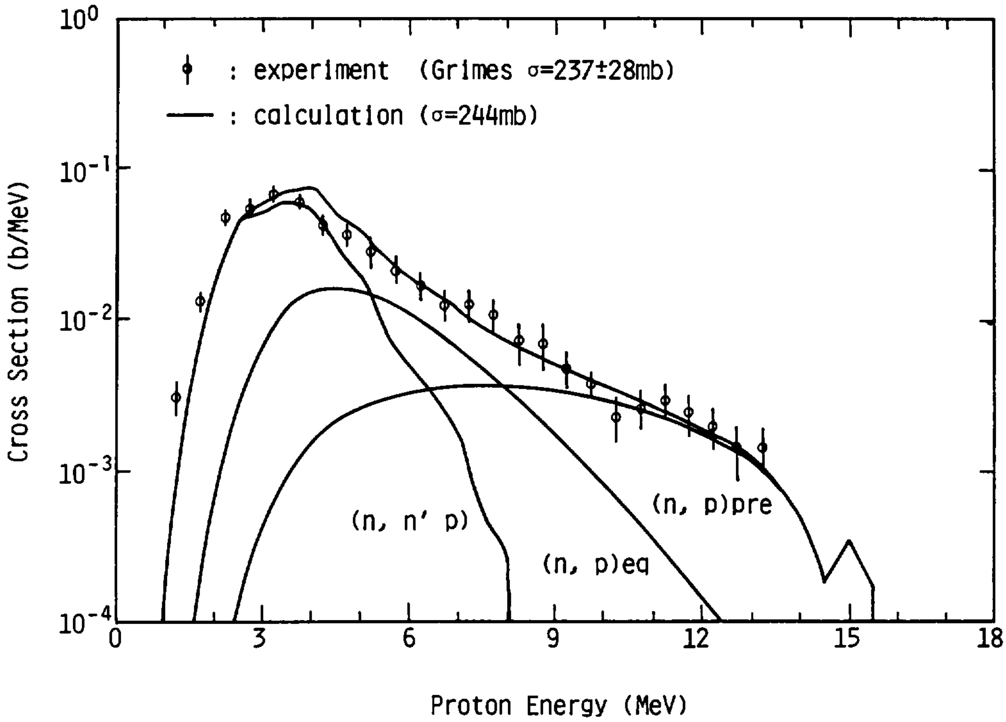

Fig. 2 Proton energy spectra from copper at En = 14.8 MeV

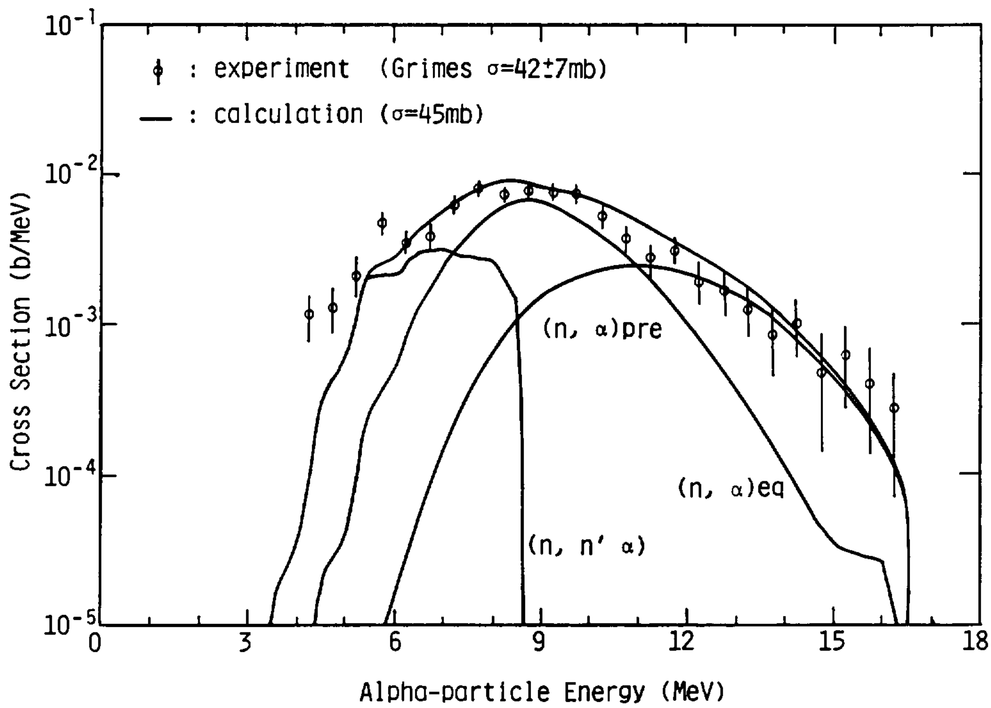

Fig. 3 Alpha-particle energy spectra from copper at En = 14.8 MeV

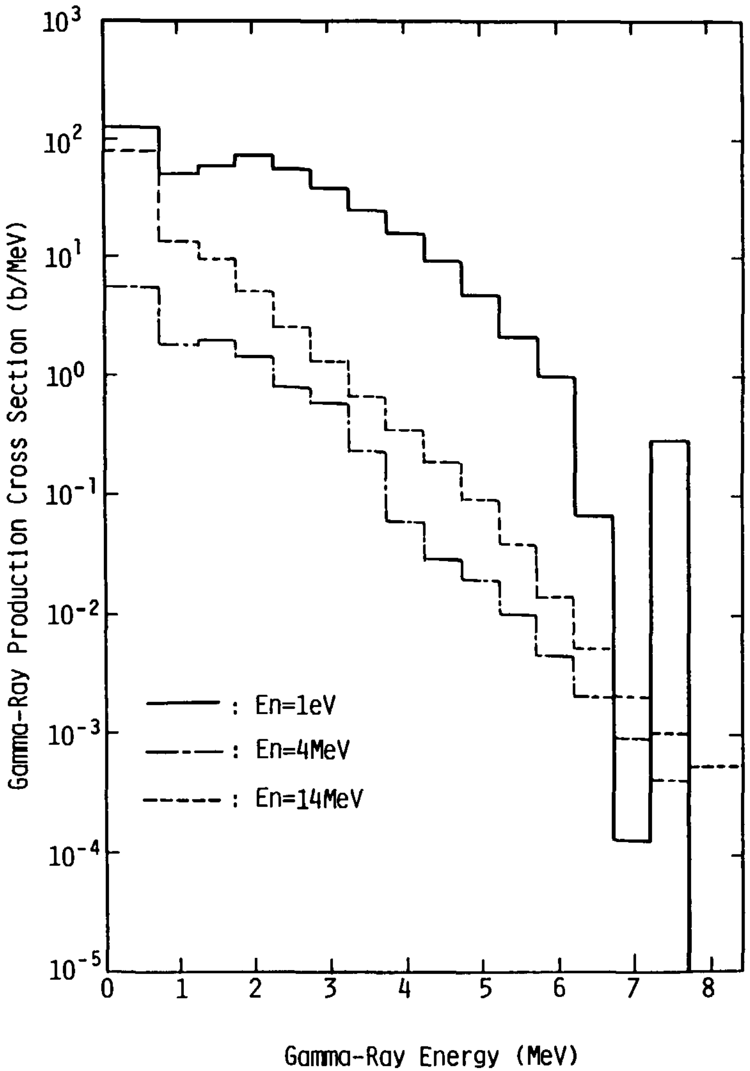

Fig. 4 Gamma-ray energy spectra from hafnium-177

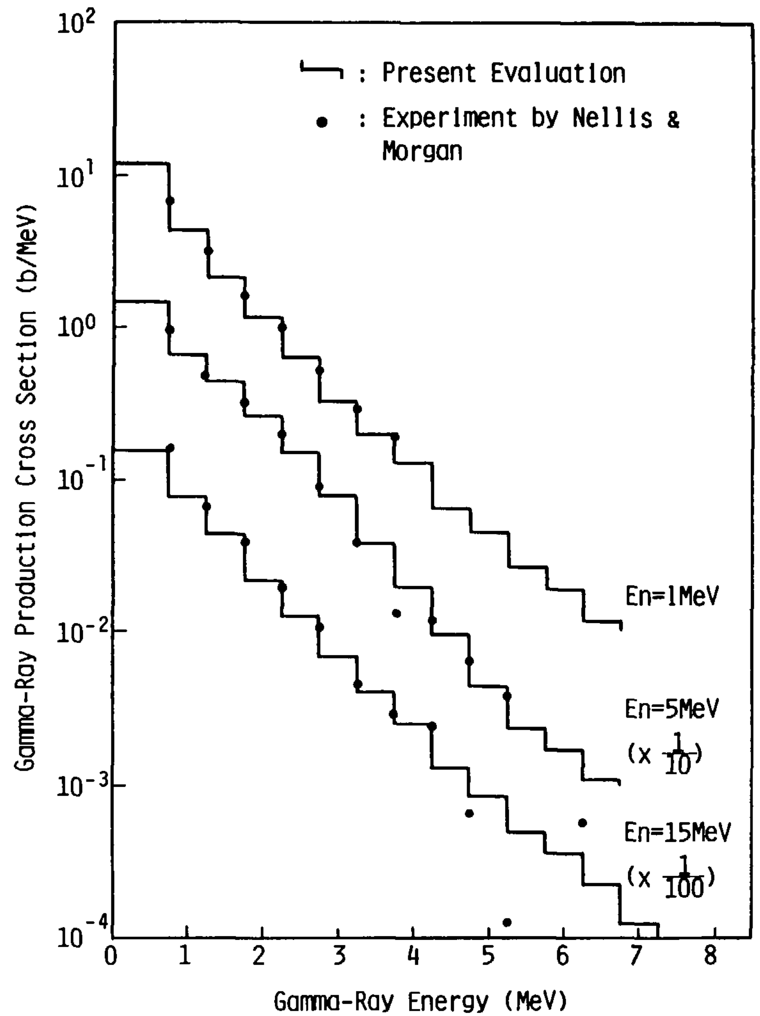

Fig. 5 Gamma-ray energy spectra from uranium-235

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VIII-1

Measurements of Double-differential Neutron Emission Cross Sections for Al, Cr, Fe and Ni

N.Yabuta, M.Baba, T.Kikuchi, M.Ishikawa and N.Hirakawa

Double-differential neutron emission cross sections were measured for aluminum, chromium, iron and nickel at the incident energy of 14.1 MeV. For iron and nickel, measurements at 18.0 MeV were also performed. From the measurements, we obtained the data of 1) neutron emission spectra and angular distribution of continuum neutrons, and 2) elastic-scattering and inelastic-scattering cross sections from the low-lying isolated levels. The results were compared with the corresponding values derived from the current evaluation.

The measurements were carried out using a time-of-flight technique at Tohoku university Dynamitron facility. Primary neutrons were produced via the T+d reaction with a Ti-T target at the emergent angle of 90° and 0° for 14.1 and 18.0 MeV measurements, respectively. Scattering samples were natural elements of right cylinder of 4-cm long and 2.5-cm (Fe and Ni) or 3.5-cm (Al and Cr) in diameter. Flight path length was ~4 or 8 m. Experimental details are described in Ref.1~3.

Typical results of the emission spectra are illustrated in Fig.1,2. The present results at 14.1 MeV were in general agreements with those at OKTAVIAN [4] except for lower energy region. The present data of emission spectra and scattering cross sections were reproduced fairly well by the recent evaluation for JENDL-3PR1; however, data comparison indicated the problems in the JENDL-3PR1, 1) overestimation of high energy continuum neutrons (i.e.,pre-equilibrium components) in iron, which is seen clearly at 18 MeV data, 2) underestimation of inelastic-scattering cross section of 3^- collective level around 4.5 MeV. Compared with ENDF/B-IV, the present results show marked disagreements for chromium and nickel, while they are in good agreements for iron in both neutron energy. For the data of aluminum, the present results are in good agreements with the ENDF/B-IV except for slight differences at backward angles. It appears that the JENDL-2 values overestimates the elastic-scattering and equilibrium process, in contrast to the underestimation in the inelastic -scattering cross sections from the low-lying levels and pre-equilibrium process.

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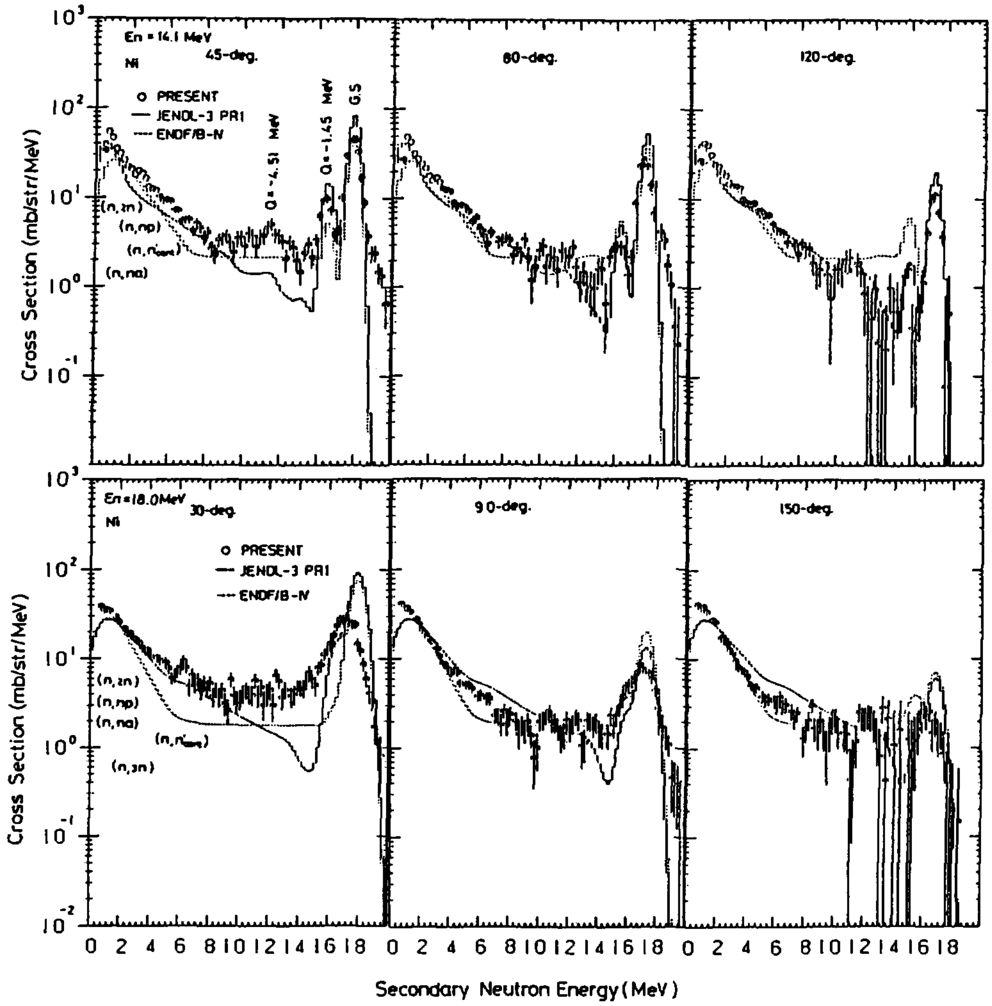

Fig. 1 Neutron emission spectra of iron.

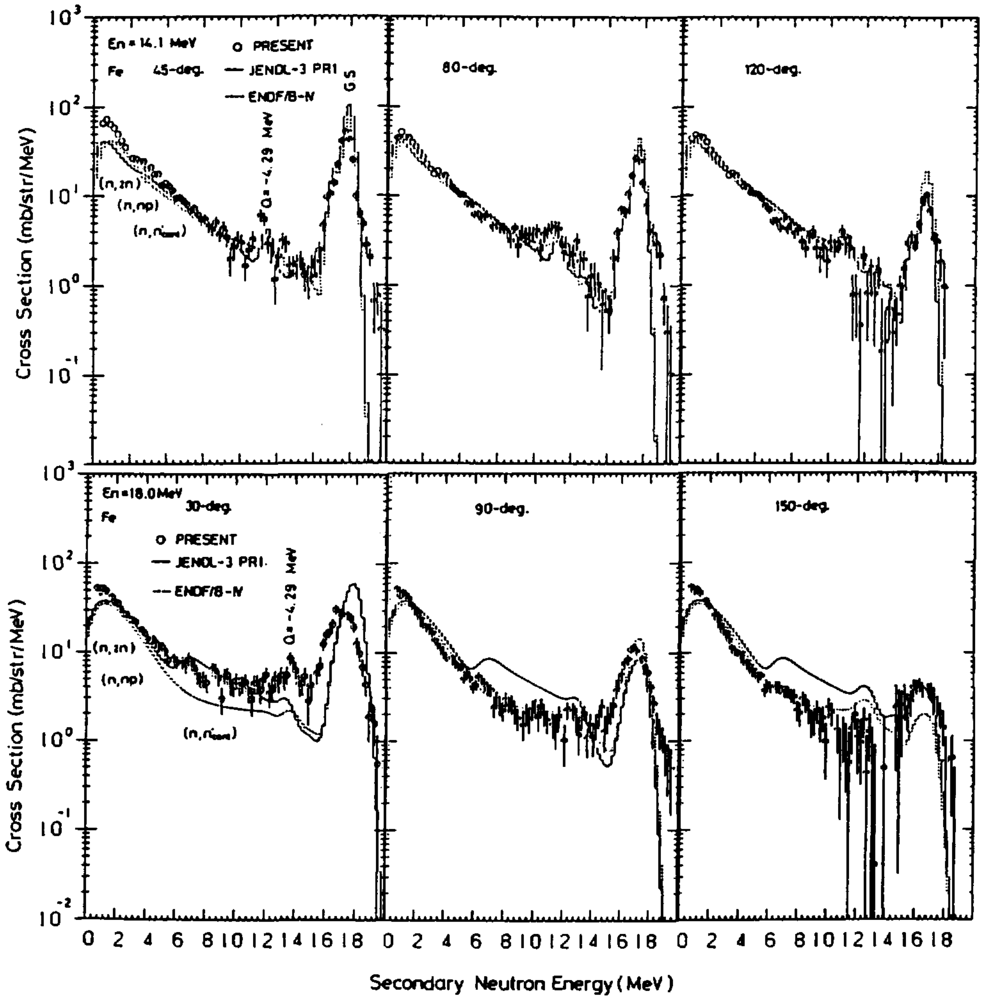

Fig.2 Neutron emission spectra of nickel.

VIII-2

Neutron Transmission Measurements for Fe, Ni and Cu in the energy range of 0.3 to 2.2 MeV.

M.Ishikawa, M.Baba, N.Yabuta, T.Kikuchi and N.Hirakawa

We have measured neutron transmission for samples of iron, nickel and copper with variety of thicknesses and derived 1) thickness-dependent total cross sections, and 2) cross sections extrapolated to zero thickness.[1] The measurements were carried out at Tohoku University Dynamitron facility using white spectrum neutrons produced by the Li(p,n) reaction on a thick Li target, and a time-of-flight technique. The transmission detector was a NE213 scintillator, 2" thick and 2" in diameter located 6.65 m from the target. The transmission samples were fabricated into right cylinders of 2-cm in diameter and 1,2,3,4 and 5 cm long. They were mounted on a eight-position sample-changer wheel which was on-lined with the T-O-F analyzer. The measurements were performed concurrently for five samples of interest with different thicknesses and two polyethylene samples, 1.65 and 5 cm long, by rotating the wheel at ~ 2rpm. The data were obtained for energy bins corresponding to the energy resolution of the spectrometer (~0.3 ns/m).

The results of polyethylene samples agreed each other and with the recommended values within 2 %. The results of iron and nickel showed thickness-dependence appreciably stronger than expected from the values by JENDL-3PR1 and ENDF/B-VI, and higher values in extrapolated cross sections by several %. Typical results of averaged cross sections are shown in Fig.1. These results are in qualitative agreements with those reported by Smith et al[2], and suggest more enhanced structure in the total cross sections. For copper, measured values did show much milder variation with sample thicknesses except for low energy region, where cross section fluctuates significantly.

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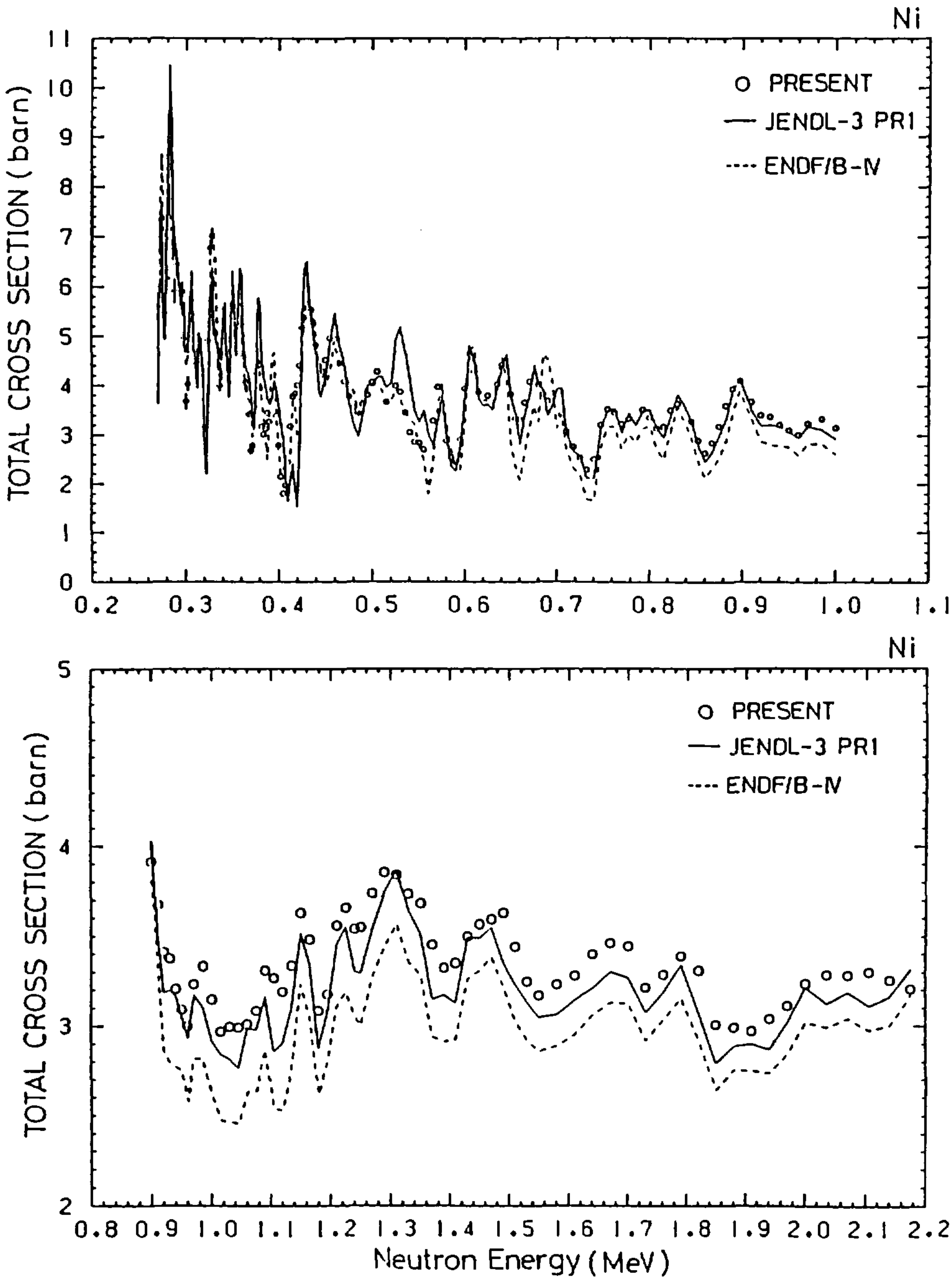

Fig. 1 Energy averaged neutron total cross sections of nickel.

VIII-3

Measurements of fission cross section ratios of U-236, Np-237 and Am-243 relative to U-235 from 0.7 to 7 MeV

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M.Baba and N.Hirakawa

Neutron induced fission cross section ratios were measured relative to U-235 for U-236, Np-237 and Am-243 in the energy range from 0.7 to 7.0 MeV. Measurements were carried out using mono-nergetic neutrons at the Dynamitron facility of Tohoku University. Experimental details were described previously.[1] Neutrons between 4.2 and 7.0 MeV were produced by the $D(d,n)^3He$ reaction with a deuterium gas target. The $T(p,n)^3He$ reaction with a metallic tritium-titanium target was chosen for the energy range from 0.7 to 3.0 MeV. The mean energies and energy spreads of the source neutrons were measured by the time-of-flight method with a NE213 scintillation detector.

Fission samples were electroplated on platinum plates of 36 mm in diameter and 0.3 or 0.4 mm thick. Each deposition (25 mm in diameter) was sintered into oxide to fix on the plate. Isotopic compositions and number of fissionable nuclides in the sample were determined by the alpha counting and thermal neutron irradiation.

The fission detector used in this study was a parallel plate gas-flow type ionization chamber. The U-235 and the other sample to be measured were placed back to back in the fission chamber. A measurement was composed of two runs where both samples were equally faced to

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the neutron target. By this method, corrections such as the neutron attenuation by the chamber and the fission fragments anisotropy could be eliminated.

The samples were irradiated at the distance of 5-10 cm from the target. Measurements of U-236 and Np-237 were performed by using continuous neutron beam. Background caused by room-returned neutrons for these beam was measured by placing an iron bar (50cm long) between the target and the chamber. For Am-243, pulsed beam was used to separate the background due to the spontaneous fission of Cm-244 existing as an impurity in this sample.

Fission cross section ratios were obtained after corrections for 1) room-returned background, 2) extrapolation to zero pulse height of the fission fragment spectrum, 3) self-absorption of the fission fragments in the sample layers, 4) impurities in the samples, 5) parasitic low energy neutrons. The experimental errors of the ratios were analyzed by taking the covariances of the measuring data into account.

The present results are shown in Fig 1-3. The following could be observed.

(a) The present U-236 results agree very well with previous data, especially with those by Behrens et al.[2]

(b) The Np-237 results are in good agreement with those by Meadows [3] and with Behrens et al[4] in shape, although the present results are about 10 % higher in magnitude.

(c) The Am-243 results agree with those of Behrens et al[5]in shape, however,the present results are lower by 20 to 30 % in magnitude.

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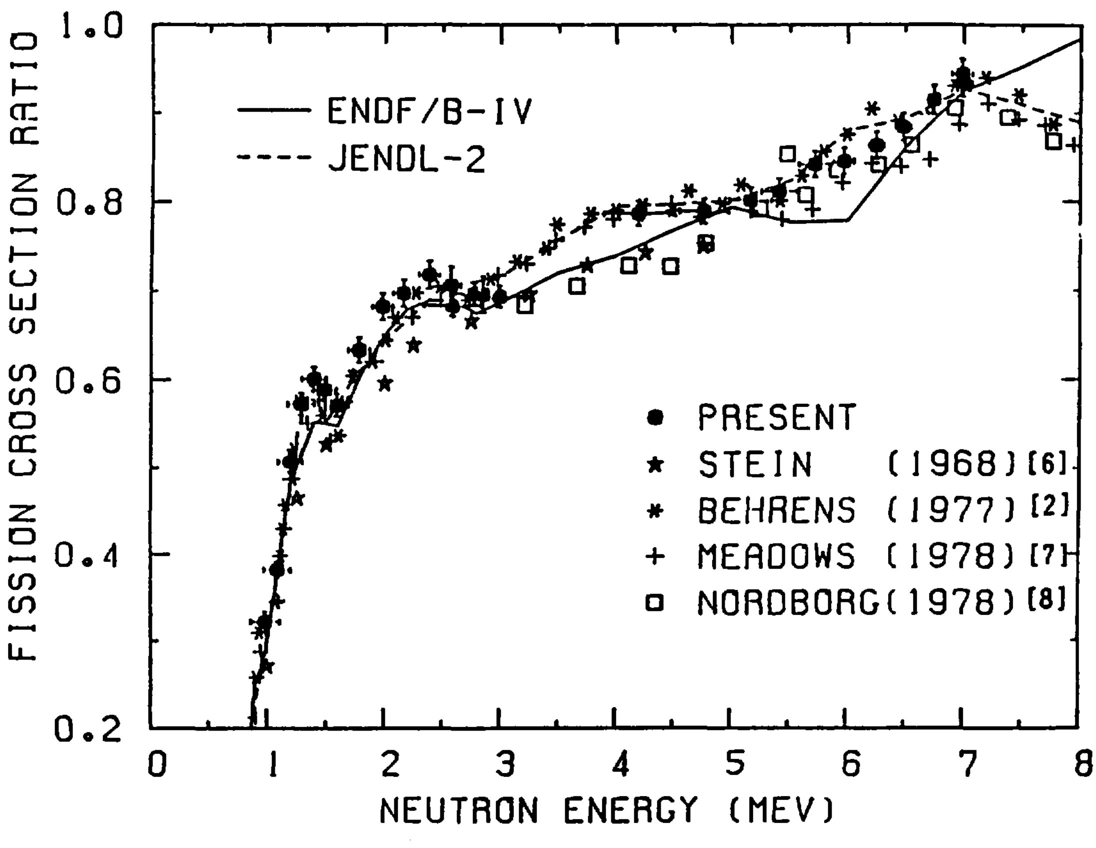

Fig. 1 Comparison of the fission cross section ratio of ^{236}U relative to ^{235}U

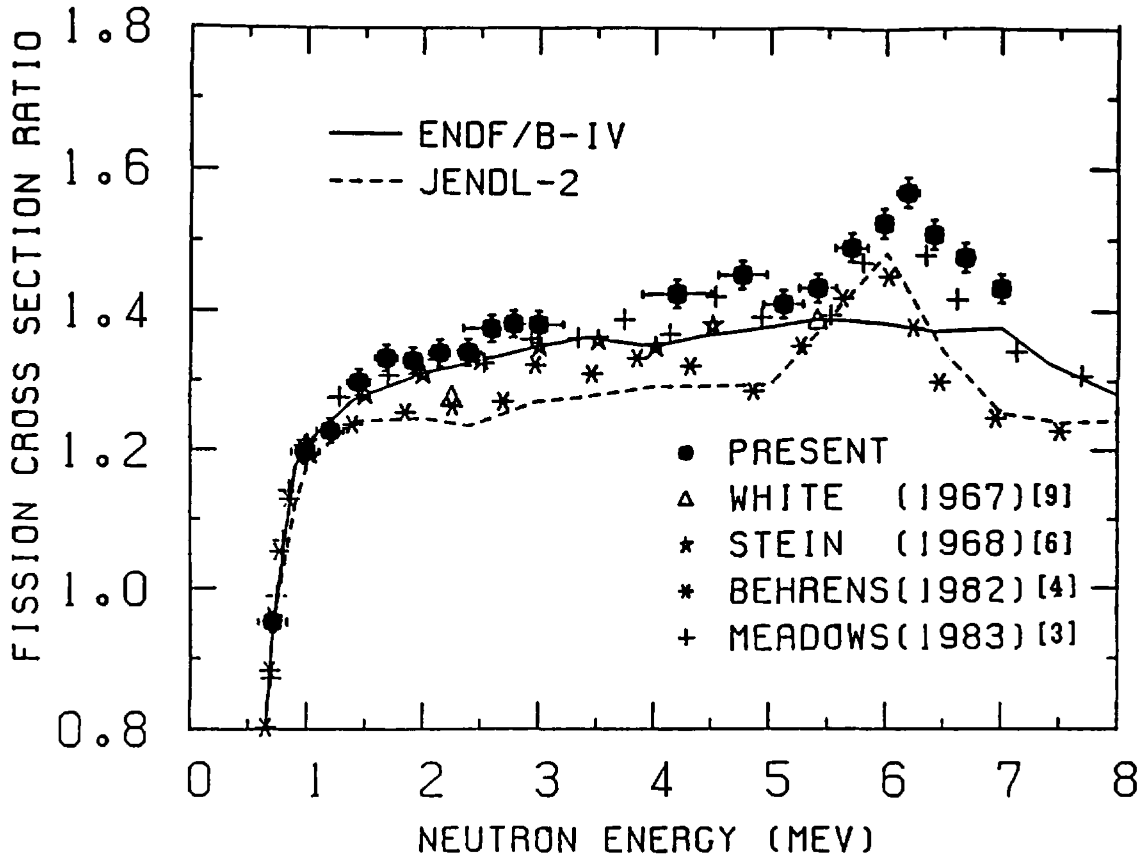

Fig. 2 Comparison of the fission cross section ratio of ^{237}Np relative to ^{235}U

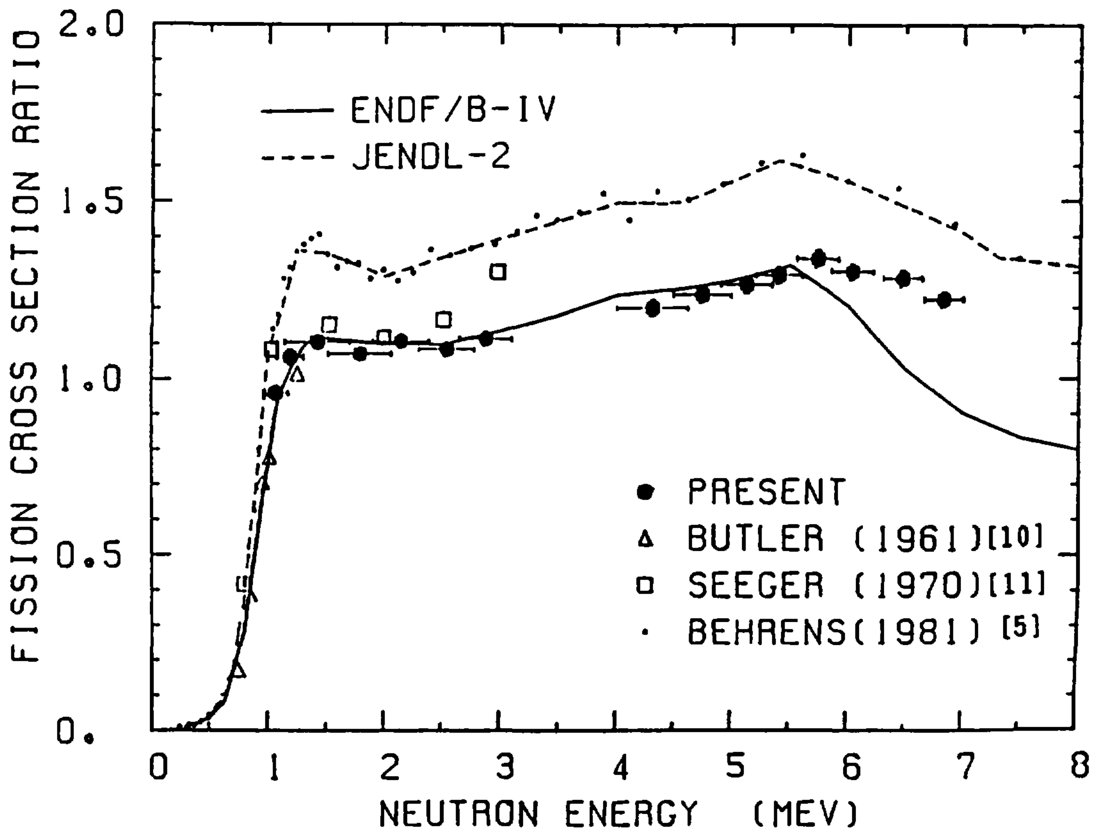

Fig. 3 Comparison of the fission cross section ratio of ^{243}Am relative to ^{235}U