# 3.3 Verification of Dosimetry Cross Sections above 10 MeV based on Measurement of Activation Reaction Rates in Fission Neutron Field

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To validate the dosimetry cross sections in fast neutron energy range, activation reaction rates were measured for 5 types of dosimetry cross sections which have sensitivity in the energy rage above 10 MeV utilizing JRR-4 reactor of JAERI. The measured reaction rates were compared with the calculations reaction rates by a continuous energy monte carlo code MVP. The calculated reaction rates were based on two dosimetry files, JENDL Dosimetry File and IRDF-90.2.

## 1. Introduction

The Japan Atomic Energy Research Institute is conducting a design study of advanced marine reactor (MRX)[1]. The MRX is an integral type PWR with a water-filled containment vessel, in-vessel type control rod drive mechanisms and an emergency decay heat removal system using natural convection. One of important issues in the shielding design of the MRX is to estimate the dose rate in vicinity of the main steam line and the coolant line due to high energy $\gamma$ rays from the secondary coolant in the steam generator which is located closed to the reactor core. Neutron transport calculation above 10 MeV plays an important role in such a dose estimation. In the shielding experiment to validate the shielding design, the activation method is one of established procedures to estimate the neutron attenuation in the shielding materials. For accurate analysis of the experiment, highly accurate dosimetry cross sections are required. In this study, dosimetry cross sections which is able to use in the energy region above 10 MeV and in the fission neutron filed were tested by comparing the measured reaction rates and calculated ones. The experiment is described in section 2 and the calculation of reaction rates is presented in section 3. In section 4, discussion regarding validity of the dosimetry cross sections in JENDL Dosimetry File[2] and IRDF-90.2[3] is described.

## 2. Measurement of Activation Reaction Rates

The JRR-4 reactor of JAERI, which is a light-water-moderated swimming pool type reactor whose nominal power is 3.5 MW, was used in the present study. The horizontal and vertical cross sections of the reactor is shown in Fig. 1. Reaction rates were measured in the pneumatic tube irradiation facility and aluminum pipe installed in water pool outside of the reactor core tank. Reaction rates were measured for 5 types of reactions listed in table 1. There exists limited number of reactions which can be used for fast neutron dosimetry above 10 MeV in fission neutron filed while there are many reactions for fusion neutron dosimetry applications. Size of the irradiated samples and other characteristics is also listed in the table 1. The reactions have high sensitivities in fast neutrons above 10 MeV except for the $^{27}Al(n,\alpha)^{24}Na$ reaction which were considered as standard dosimetry reaction. Irradiation time was 1 to 2 minutes in pneumatic tube and 1 to 2 hours in Al pipe. The samples were irradiated when the power of the reactor was 3.5 MW. After the irradiation, induced $\gamma$ rays were detected by high purity Ge detector. The detector efficiency was determined by experiment using standard $\gamma$ ray sources. Distance from the detector surface to the sample was set to 20 cm to eliminate the sum-coincidence effect. As a correction factor, $\gamma$ ray attenuations in sample itself was taken into account by one dimensional approximation. Correction for perturbation of neutron flux during the irradiation was neglected. Measured reaction rates are shown in table 2. The uncertainties of the reaction rates were mainly attributed to the uncertainty in determination of detector efficiency which were considered to be 5%.

3. Calculation of Activation Reaction Rates

To validate the dosimetry cross sections, calculated reaction rates and measured ones were compared. The vectorized continuous energy Monte Carlo code MVP[4], running on vector and parallel supercomputer VPP500 of JAERI, was used for the calculation of the reaction rates. The nuclear data library used in the code was prepared from JENDL-3.2. Three dimensional geometry was modeled in the MVP calculation as shown in Fig.1. The track length estimator was used to estimate the neutron flux in pneumatic pipe and Al pipe. Larger detector regions than those of actual irradiated sample were taken for efficient scoring of track length. Gradation of the neutron flux was considered to be very small because the detector region were filled with air or nitrogen gas. The weight window sampling method was employed as variance reduction technique. Neutrons with energy above 3 MeV was generated uniformly in reactor core region as input neutron source. The neutron energy distribution of the source was taken from that of $^{235}U$ in JENDL-3.2[5].

Calculated neutron spectra in pneumatic tube and Al pipe are shown in Fig. 2. The 90% response ranges of the dosimetry reactions for $^{235}U$ fission neutron spectrum are also shown in the figure. For the result of pneumatic tube, the neutron flux smoothly decrease as increase the neutron energy while several structures can be found in spectrum in Al pipe. The fractional standard deviation in each energy bin is less than 0.05 for both of the calculations.

4. Discussion

Comparison between measured reaction rates and calculated ones was shown in table 3. Because the uncertainties of calculated reaction rates is small (0.3 to 3 %), uncertainties of the C/E values are close to the ones of measured reaction rates. Calculated reaction rates were agreed with experimental data within 10 to 20 % except for the $^{127}I(n,2n)^{126}I$ reaction calculated by JENDL Dosimetry File whose discrepancy were 32% in the pneumatic pipe and 31% in Al pipe. The discrepancy can be improved if the cross section in IRDF-90.2 is used. Apparent difference between cross sections in the JENDL Dosimetry File and the IRDF-90.2 can be seen in Fig.3 which shows the differential data in both the files and experimental data[6-19] retrieved from the EXFOR[20]. Reason of the overestimation of reaction rate by JENDL Dosimetry File is attributed to the large cross section values from threshold energy to 16 MeV. Re-evaluation of the JENDL Dosimetry File is recommended to use this reaction in fast neutron dosimetry in fission neutron filed.

5. Conclusion

Activation reaction rates were measured for reactions which have sensitivities in the energy range above 10 MeV. The measured reaction rates were compared with calculated ones deduced from monte carlo calculation by MVP code. Calculated data were agreed with experimental data within 10 to 20 % except for the $^{127}I(n,2n)^{126}I$ reaction from JENDL Dosimetry File. From the comparison of measured data and calculated ones, $^{93}Nb(n,2n)^{92m}Nb$, $^{90}Zr(n,2n)^{89}Zr$ and $^{58}Ni(n,2n)^{57}Ni$ reaction cross sections in JENDL Dosimetry File and IRDF-90.2 are accurate enough for the neutron dosimetry for shielding experiment we concern. Revision of the $^{127}I(n,2n)^{126}I$ reaction cross section in JENDL Dosimetry File is required so that the reaction is useful for fast neutron dosimetry above 10 MeV in fission neutron filed.

## 6. Acknowledgements

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## 7. References

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[20] EXFOR, experimental data base available from OECD NEA Data Bank.

Table 1. Characteristics of reactions used for measurement of activation reaction rates. (*90% response for the $^{235}U$ fission spectrum. **$^{235}U$ fission spectrum averaged cross section.)

| Reaction | Threshold energy (MeV) | Response* (MeV) | Cross Section (mb)** | Half Life | γ-ray Energy(keV), branching ratio | Size of the Detector (mm) |
|---|---|---|---|---|---|---|
| $^{27}Al(n,\alpha)^{24}Na$ | 3.2 | 6.4 - 12.0 | $7.20\times10^{-1}$ | 14.959h | 1368.63(99.99) | 10×10×0.5t |
| $^{93}Nb(n,2n)^{92m}Nb$ | 8.9 | 9.4 - 15.0 | $4.80\times10^{-1}$ | 10.15d | 934.53(100.0) | 5φ3t |
| $^{127}I(n,2n)^{126}I$ | 9.2 | 9.7 - 15.0 | 1.05 | 13.02d | 666.33(33.1) | 10φ5t |
| $^{90}Zr(n,2n)^{89}Zr$ | 12.1 | 13.0 - 19.0 | $1.03\times10^{-1}$ | 3.268d | 909.15(99.01) | 5φ3t |
| $^{58}Ni(n,2n)^{57}Ni$ | 12.4 | 12.5 - 19.2 | $3.04\times10^{-3}$ | 35.9h | 1377.59(80.0) | 10φ5t |

Table 2. Measured reaction rates in the pneumatic tube (upper column) and in the Al pipe (lower column) with the uncertainties (shown in parenthesis).

| Reaction | Measured Reaction Rate ($sec^{-1}\cdot watt^{-1}$) |
|---|---|
| $^{27}Al(n,\alpha)^{24}Na$ | $1.17\times10^{-21}$ (5.0%) |
| | $1.47\times10^{-21}$ (5.2%) |
| $^{93}Nb(n,2n)^{92m}Nb$ | $7.31\times10^{-22}$ (5.4%) |
| | $1.05\times10^{-23}$ (5.3%) |
| $^{127}I(n,2n)^{126}I$ | $2.11\times10^{-21}$ (5.3%) |
| | $2.98\times10^{-23}$ (5.5%) |
| $^{90}Zr(n,2n)^{89}Zr$ | $1.80\times10^{-22}$ (6.1%) |
| | $3.19\times10^{-24}$ (6.1%) |
| $^{58}Ni(n,2n)^{57}Ni$ | — |
| | $1.11\times10^{-25}$ (13%) |

Table 3. The C/E values of reaction rates calculated from JENDL Dosimetry File and IRDF-90.2. Figures in upper column show the C/E values for the pneumatic tube while the those in lower column show the C/E values for the Al pipe.

| Reaction | JENDL Dosimetry File | IRDF-90.2 |
|---|---|---|
| $^{27}Al(n,\alpha)^{24}Na$ | 0.977 | 1.02 |
| | 1.12 | 1.17 |
| $^{93}Nb(n,2n)^{92m}Nb$ | 1.14 | 1.06 |
| | 1.11 | 1.03 |
| $^{127}I(n,2n)^{126}I$ | 1.32 | 0.966 |
| | 1.31 | 0.986 |
| $^{90}Zr(n,2)^{89}Zr$ | 0.944 | 0.903 |
| | 0.821 | 0.790 |
| $^{58}Ni(n,2n)^{57}Ni$ | - | - |
| | 0.862 | 0.900 |

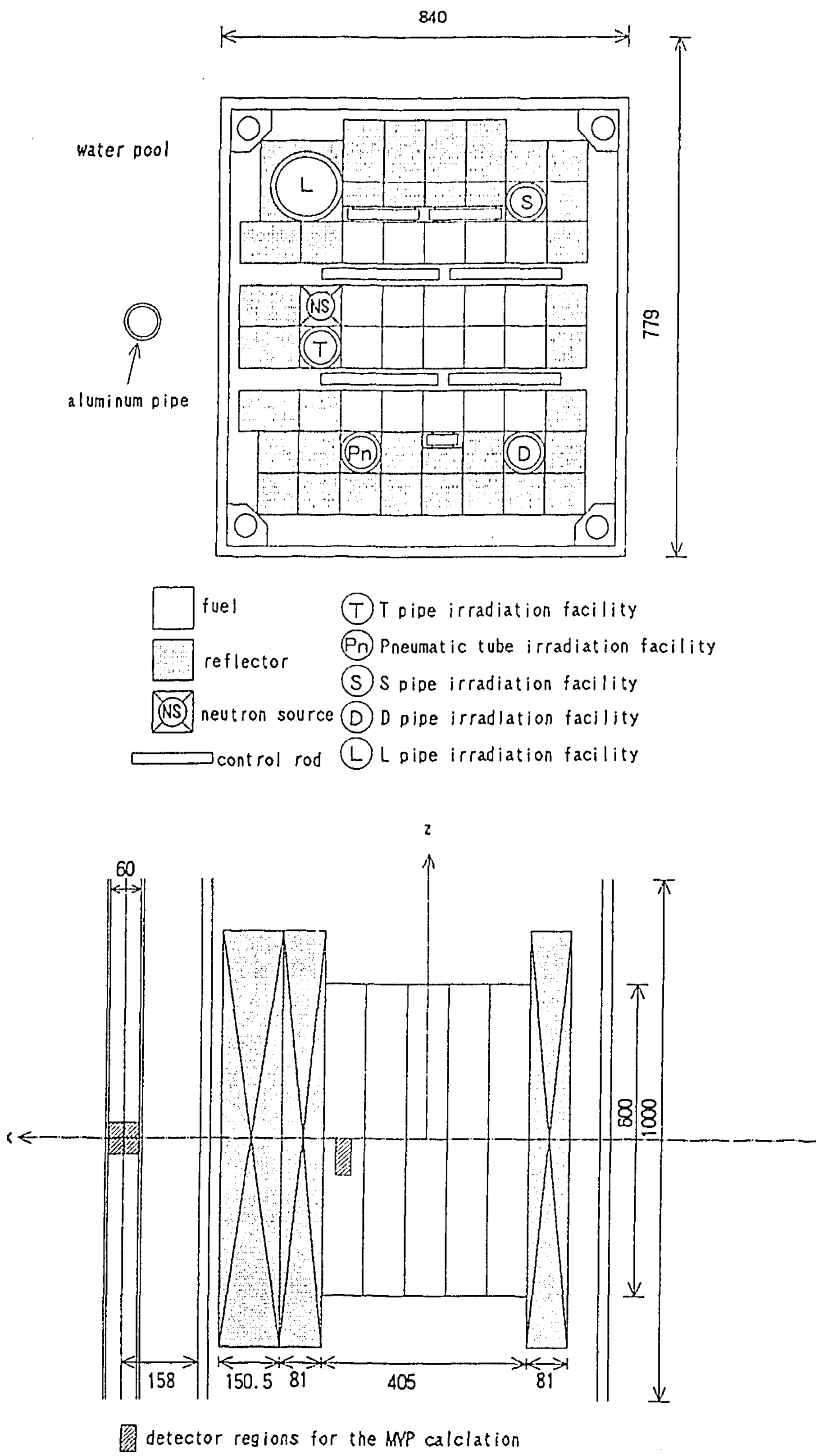


Fig. 1. Cross sectional view of the JRR-4 reactor core (unit in mm). Upper figure shows horizontal cross section and lower one shows vertical cross section.

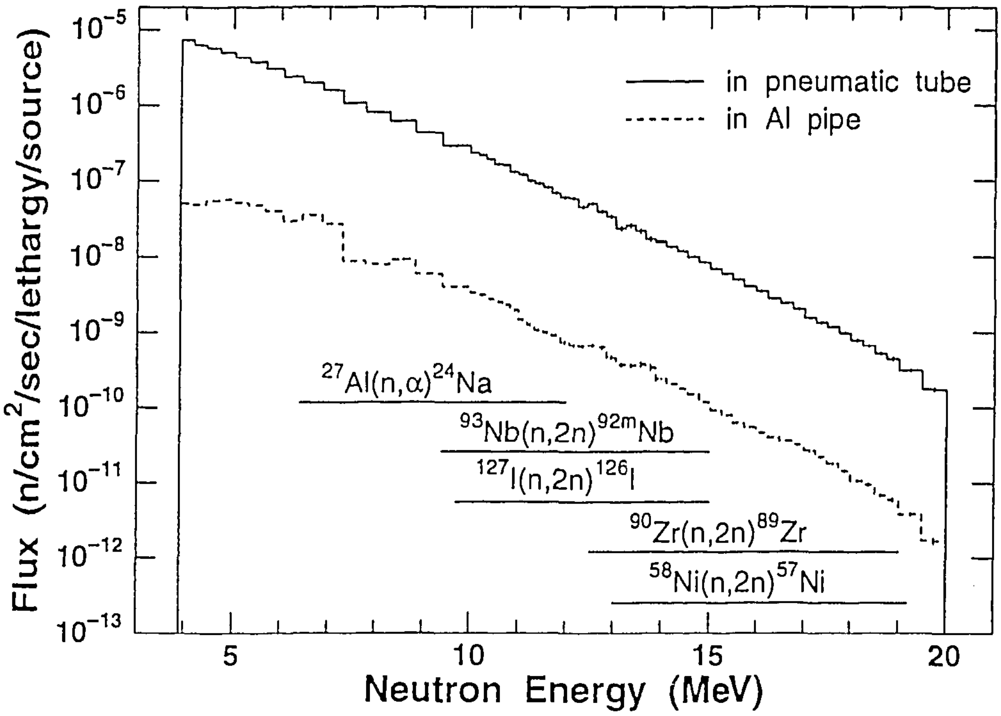


Fig. 2. Calculated neutron spectra in the pneumatic tube and Al pipe using the MVP code. Solid line shows the result for pneumatic tube while dashed line indicates the result for Al pipe. The 90% response ranges for dosimetry cross sections are also shown in the figure.

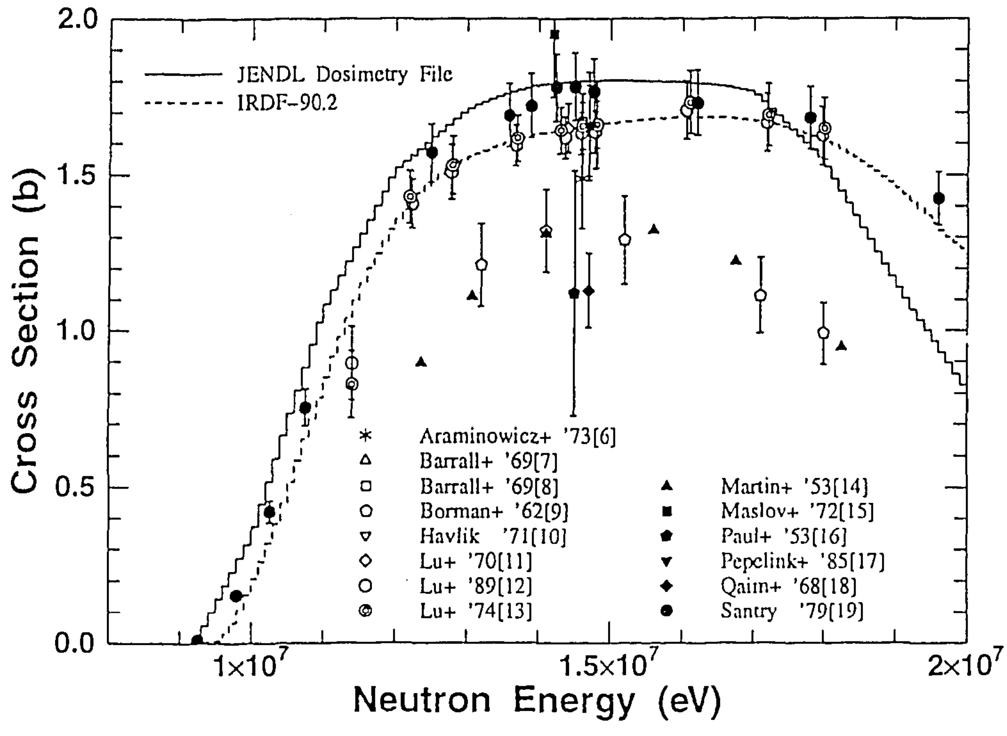


Fig. 3. Comparison of $^{127}I(n,2n)^{126}I$ reaction cross sections in the JENDL Dosimetry File and the IRDF-90.2 with experimental data from EXFOR, experimental data base available from OECD NEA Data Bank.