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July 1973

日本原子力研究所
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JAERI レポート

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JAERI Report

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編集兼発行　日本原子力研究所

印　　刷　いばらき印刷

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Nuclear Accident Dosimetry System in JAERI
(with Experiments of Simulated Criticality Accident)

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Received 27 January 1973

Abstract

A new dosimetry system for nuclear accidents is designed in JAERI. It is composed of six fission track detectors (UO_2 or ThO_2 doped glasses) and one sulfur disc.

Since it is necessary for the assessment of neutron doses in nuclear accidents to know the shape of neutron spectra, it is assumed that the spectra of fast, intermediate and thermal neutrons in nuclear accidents are expressed in the form of $\frac{2}{\sqrt{\pi}}\sqrt{a^3 E}\exp(-aE)$, $E^n/I(n)$ and Maxwellian distribution function, respectively, where E is neutron energy in MeV, and n are parameters which specify the shape of spectra of fast and intermediate neutrons, respectively, and $I(n)$ is a normalization factor for the spectra of intermediate neutrons.

The theory of determination of the parameters (a and n) and the fluences of fast, intermediate and thermal neutrons with this system is described in detail.

The system was exposed, in practice, to neutrons of simulated criticality accidents. It was possible to determine the spectra of the fast neutrons as well as the fluences of fast and thermal neutrons. However, it was impossible to decide the spectra of intermediate neutrons. The reasons are discussed.

The detection limit of this system is about 50 rems (as ICRP dose equivalent) and 10 rads (as surface absorbed dose).

原研型核事故線量測定器

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（1973年1月27日受理）

要　　旨

エネルギー・スペクトルが解らないと中性子の線量を求めることは困難である。中性子のスペクトルを三つの成分に分け，速中性子のスペクトルを$\frac{2}{\sqrt{\pi}}\sqrt{a^3 \mathrm{E}}\ \exp(-a\mathrm{E})$で表わし，中速中性子は$\mathrm{E}^n$分布，熱中性子はMaxwell分布するものとした。ここで，aおよびnはエネルギースペクトルのパラメータであって，スペクトルの形を決定する因子である。パラメータa，nを，天然ウラン，微濃縮ウランまたはThをドープしたガラス製核分裂検出器6個と硫黄ディスク1個で構成された原研型検出器系を用いて求める方法について述べた。また，入射中性子流量，線量を評価する方法も述べた。

この原研型検出器系を用い，JRR－4のウラン・コンバータから発生する核分裂中性子のエネルギースペクトルを求め，線量を評価する実験を行なった。速中性子成分のみの平均エネルギーは約1.7MeVであった。非減速の核分裂中性子の平均エネルギーは約2MeVであるから，若干減速していることが解った。中性子成分のパラメータnは求めることができなかったので，その原因についても考察した。

Contents

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1. Introduction

JAERI has participated in the fifth nuclear accident dosimetry intercomparison study arranged at Oak Ridge National Laboratory in 1968[1]. The doses evaluated by means of the dosimetry system of JAERI were a little lower than the average values of the doses reported by other participants, but were not the lowest. The weak points of the dosimetry system of JAERI at that time were that the dosimeters were not used routinely and that the dose assessment could not be done as quickly as other participants.

Recently, some improvements for the dosimetry system have been carried out. In this report an explanation of a new nuclear accident dosimetry system in JAERI is given in Sections 2, 3 and 4 and the results of dosimetry experiments on simulated criticality accidents performed in JAERI is described in Sections 5 and 6.

2. Energy Spectrum and Parameter

It is convenient for the assessment of neutron dose received by a person to express the energy spectrum of the neutrons by an appropriate formula.

In nuclear accident dosimetry, it is sometimes assumed that the neutron spectrum above Cd cut-off energy is expressed by two components, that is, (1) an intermediate components, ϕ_{in}, which is explained as a spectrum of slowing down neutrons proportional to E^{-1} at the range of from 0.5×10^{-6} to 1.0 MeV (E is energy of neutrons) and (2) a fast neutron components, ϕ_f, which is explained as an uncollided fission neutron spectrum, and has a formula represented in the following way:

$$\phi_f = \text{const} \cdot \sinh(2.29\,E)^{\frac{1}{2}} \cdot \exp(-E/0.965). \tag{1}$$

This assumption is valid for the well slowing down spectrum such as that of thermal reactors[2] as shown in **Fig. 1.** However, in the case of non-slowing down spectrum such as Assembly Godiva (LASL)[3], the assumption is not valid as indicated in **Fig. 2**, and the discrepancy between the real spectrum and the uncollided fission spectrum is not made up by a $1/E$ spectrum.

Therefore, in this study, the following neutron spectra are assumed. That is,

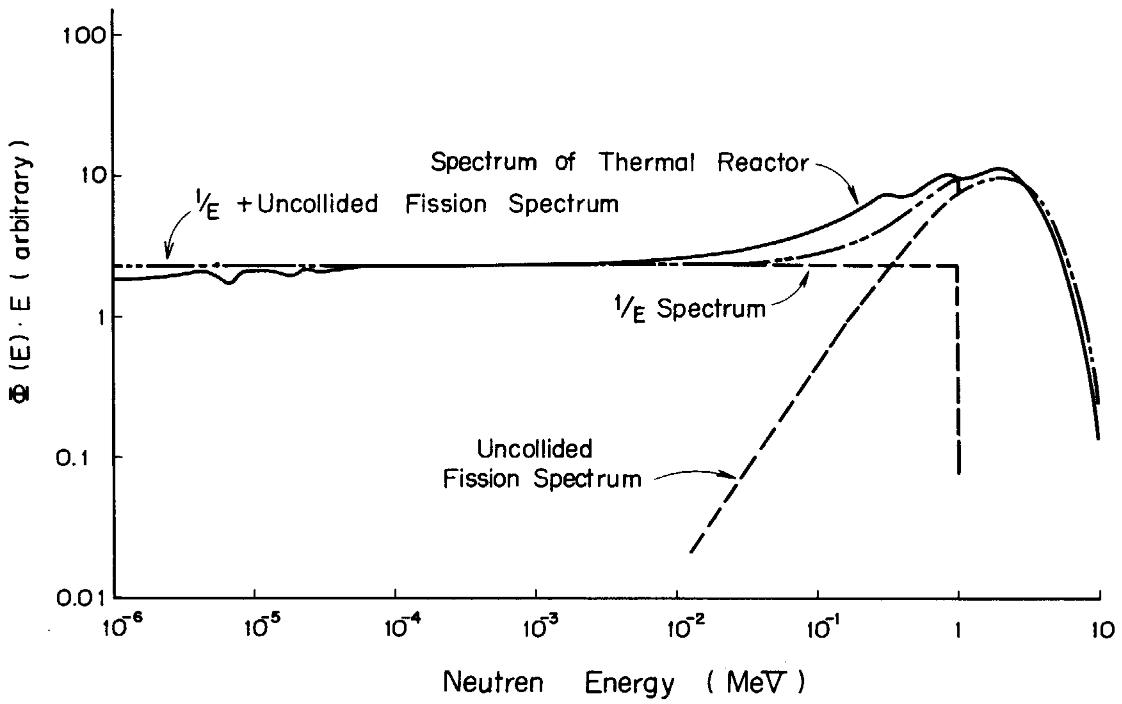

Fig. 1 Comparison between a well thermalized spectrum and a $1/E$ distribution plus an uncollided fission spectrum.

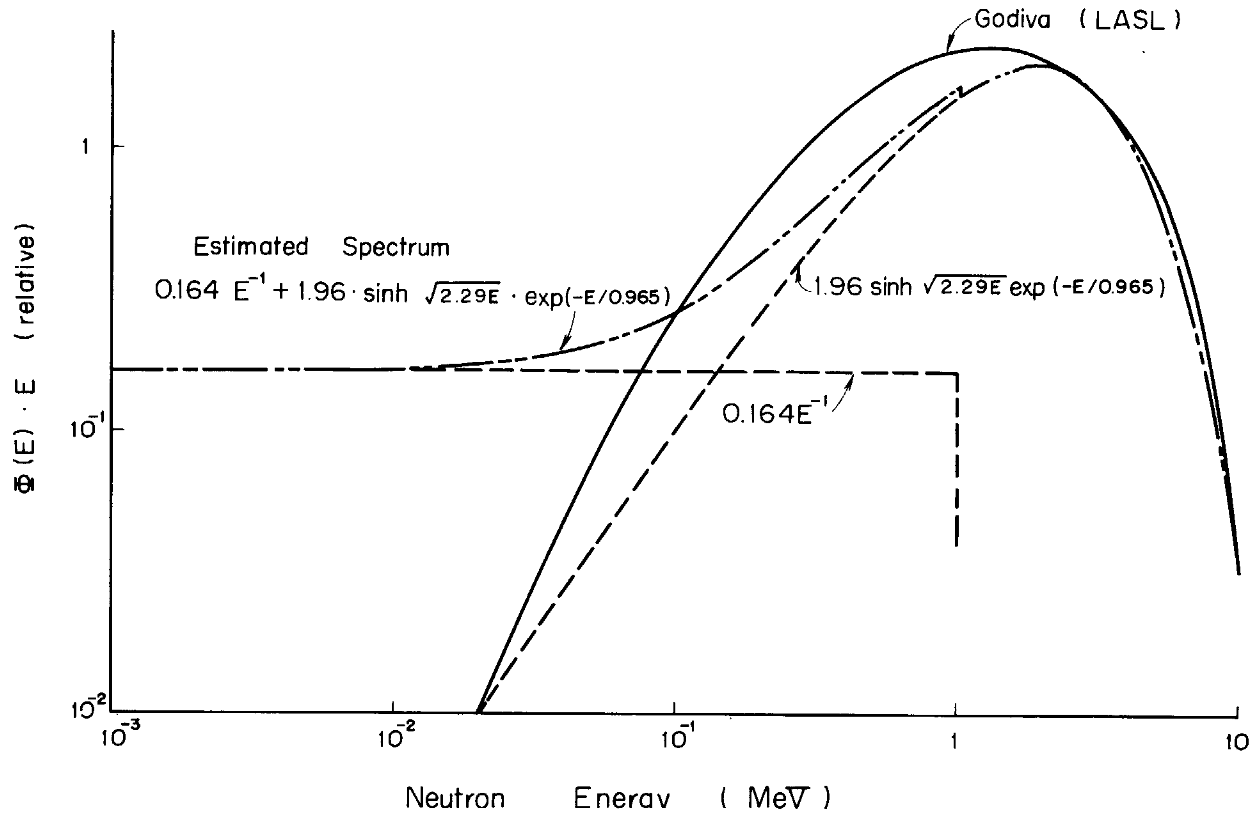

Fig. 2 Comparison between the spectrum of Godiva* and a $1/E$ distribution plus an uncollided fission spectrum. (* after Dennis, J.A.[3])

$$\phi_{in}(n, E)\,dE = \frac{E^n}{I(n)}\,dE \quad , \quad 0.5\times 10^{-6} \leq E \leq 1.0 \text{ (MeV)} \tag{2}$$

and

$$\phi_f(a, E)\,dE = \frac{2}{\sqrt{\pi}}\sqrt{a^3 E} \cdot \exp(-aE)dE \tag{3}$$

where $\phi_{in}(n, E)$ and $\phi_f(a, E)$ are normalized spectra of intermediate and fast neutrons, respectively, n and a are constants and $I(n)$ is a normalization factor to unity depending on n. In this paper, the exponent a and the coefficient n are called (energy spectrum) parameter. In this formulation, when the parameter n is minus one, $\phi_{in}(-1, E)$ is equal to $1/E$-distribution, and when the parameter a is 0.776, $\phi_f(0.776, E)$ is almost equal to the uncollided spectrum[4]. Since the average neutron energy of $\phi_f(a, E)$ is expressed as $3/(2a)$MeV, the larger the parameter a is, the smaller the average energy is.

A total neutron fluence spectrum $N_{total}\,\phi_{total}(E)$, is given by a linear combination of these spectra and a Maxwellian distribution for thermal neutrons. That is,

$$N_{total}\phi_{total}(E)dE = \left\{ N_{th}\phi_{th}(E) + N_{in}\phi_{in}(n,E) + N_f\phi_f(a, E) \right\} dE \tag{4}$$

where N_{th}, N_{in} and N_f are the thermal, intermediate and fast neutron fluences, respectively, and $\phi_{total}(E)$ and $\phi_{th}(E)$ are normalized spectra of total neutrons and the neutrons lower than Cd cut-off energy, respectively.

3. Detector System

In JAERI, fission track detectors, composed of UO_2 or ThO_2 doped glass plates[5] and sulfur discs, are used for neutron dosimetry in nuclear accidents.

3.1 Detectors for fast neutrons

The energy spectrum parameter a which specifies the shape of fast neutron spectra can be determined by appropriate combinations of ^{238}U, ^{232}Th or ^{237}Np doped glass and sulfur disc detectors. The determination method is shown, in detail, in the following section.

3.2 Detectors for thermal and intermediate neutrons

Incident neutrons are slowed down and thermalized in the body (or phantom), and the thermalized neutrons are reflected out from the body. The fraction of incident neutron fluence reflected as thermal neutrons from the phantom is rather large, especially in the region of thermal and intermediate energies, as shown in **Fig.3**[6]. Since the fission cross section of ^{235}U for thermal neutrons is very large, an enriched uranium doped glass detector was used for the estimation of thermal neutron fluence reflected from the body. One slightly enriched uranium doped glass detector and three natural uranium doped glass detectors with different filters make one set for the detection of direct incident neutrons as well as thermal neutrons reflected from the body. As shown in **Fig. 4**, one of the natural uranium doped glass fission track detectors is wholly covered with a cadmium filter whose thickness is 0.5 mm (this detector is named NU(Cd-Cd)) and two other natural uranium doped glass detectors are partially covered with cadmium filters, that is, one is not covered with cadmium but with tin for the phantom side [NU(Cd-Sn)], and the other is covered with tin for the incident side [NU(Sn-Cd)]. A tin filter is used for the protection of the glass plate against mechanical shock and for the protection of the human body by shielding α- and β- rays from the uranium. The cadmium filter also serves for the same protection purposes. The thickness of tin filters is 0.5 mm. Directly incident thermal neutron fluence is evaluated from the difference in numbers of fission tracks between NU(Sn-Cd) and NU(Cd-Cd). On the other hand, thermal neutrons reflected from the body, can be evaluated from the difference in numbers of tracks between NU(Cd-Sn) and NU(Cd-Cd).

The slightly enriched uranium doped glass fission track detector is covered with a cadmium filter wholly and this is EU(Cd-Cd). When the weight percent of ^{235}U in the slightly enriched uranium is 1.44% (twice as large as that of natural uranium), the sensitivity of EU(Cd-Cd) to intermediate neutrons also are twice as good as that of NU(Cd-Cd). Nevertheless, since the decreasing of ^{238}U content in EU(Cd-Cd) is negligible (a percentage of ^{238}U decreased from 99.28% to 98.56%), the sensitivity of EU(Cd-Cd) to fast neutrons whose energy is greater than 1.5 MeV, is almost equal to that of NU(Cd-Cd). The difference in numbers of fission tracks between EU(Cd-Cd) and NU(Cd-Cd) is due to the difference of the sensitivities to intermediate neutrons between EU(Cd-Cd) and NU(Cd-Cd) (thermal neutrons are excluded by cadmium filters). Thus, intermediate neutron fluence can

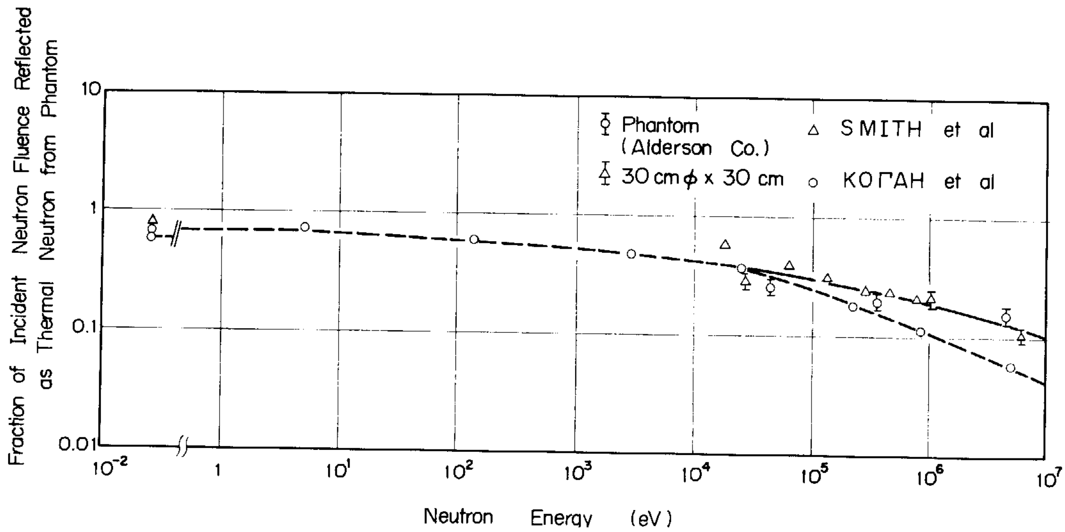

Fig. 3 Fraction of incident neutron fluence reflected as thermal neutrons from a phantom. (after Tatsuta, H. et al.[6])

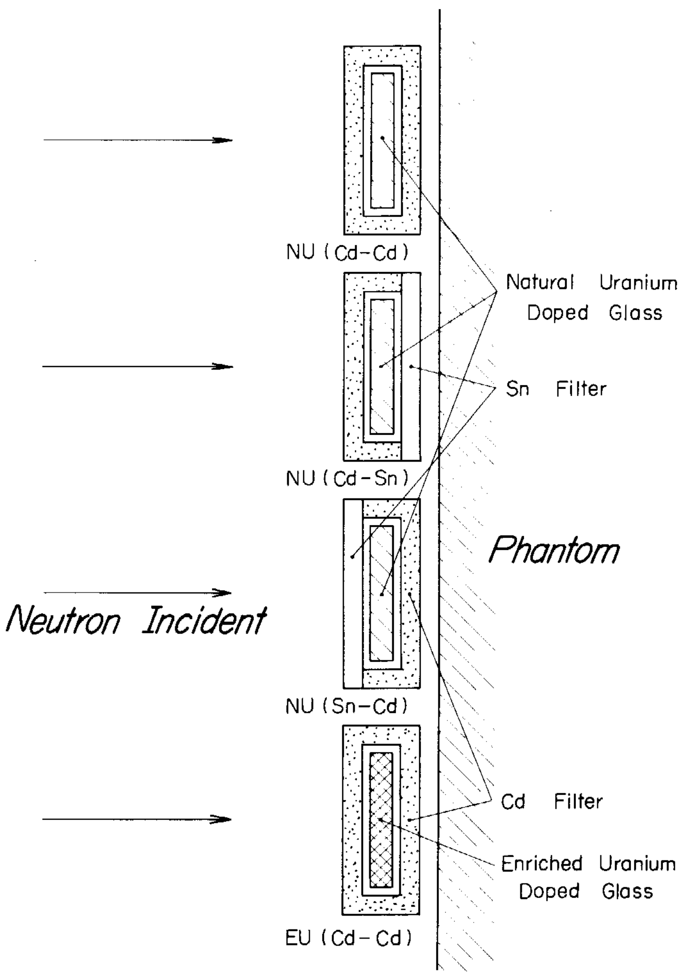

Fig. 4 Four detectors system for the estimation of thermal and intermediate neutron dose.

be detected from the combination of EU(Cd-Cd) and NU(Cd-Cd).

4. Estimation of Energy Spectrum Parameter

4.1 Estimation of parameter a

Responses of detectors, that is, number of fission tracks per unit area of ^{232}Th doped glass detectors and counting rate (cpm) of a sulfur disc are given as follows:

$$T(a, {}^{232}\mathrm{Th}) = N_f \alpha({}^{232}\mathrm{Th}) \int_0^\infty \phi_f(a, E)\, \sigma_{^{232}\mathrm{Th}}(E)\, dE$$

$$= N_f \alpha({}^{232}\mathrm{Th})\, \sigma_{\mathrm{eff}}(a, {}^{232}\mathrm{Th}) \;, \tag{5}$$

$$T(a, S) = N_f \alpha(S) \int_0^\infty \phi_f(a, E)\, \sigma_S(E)\, dE$$

$$= N_f \alpha(S)\, \sigma_{\mathrm{eff}}(a, S) \;, \tag{6}$$

where T is the number of tracks per unit area or the counting rate, $\sigma_{^{232}\mathrm{Th}}(E)$ and $\sigma_S(E)$ are cross sections of ^{232}Th and S atoms for nuclear reactions with neutrons of E(MeV), respectively, $\alpha({}^{232}\mathrm{Th})$ and $\alpha(S)$ are sensitivities (tracks/cm^2 or cpm per fluence•barn) of the detectors, which are determined using monoenergetic neutrons or neutron sources whose energy spectra are well known, and $\sigma_{\mathrm{eff}}(a, {}^{232}\mathrm{Th})$ and $\sigma_{\mathrm{eff}}(a, S)$ are effective cross sections of ^{232}Th and sulfur for neutrons whose energy spectrum is expressed as $\phi_f(a, E)$, respectively. The effective fission or activation cross section of detector depends on the shape of the spectrum or on the parameter a. These are shown in **Fig. 5**.

The ratio of the number of fission tracks per cm^2 of the ThO doped glass detector to the counting rate of the sulfur-disc becomes as follows:

$$\frac{T(a, {}^{232}\mathrm{Th})}{T(a, S)} = \frac{N_f \alpha({}^{232}\mathrm{Th})\, \sigma_{\mathrm{eff}}(a, {}^{232}\mathrm{Th})}{N_f \alpha(S)\, \sigma_{\mathrm{eff}}(a, S)} = \frac{\alpha({}^{232}\mathrm{Th})\, \sigma_{\mathrm{eff}}(a, {}^{232}\mathrm{Th})}{\alpha(S)\, \sigma_{\mathrm{eff}}(a, S)} \tag{7}$$

Since the number of tracks or counting rate, i.e., $T(a, {}^{232}\mathrm{Th})$, $T(a, S)$ and sensitivities, $\alpha({}^{232}\mathrm{Th})$ and $\alpha(S)$, are obtained experimentally, the ratio of $\sigma_{\mathrm{eff}}(a, {}^{232}\mathrm{Th})$ to $\sigma_{\mathrm{eff}}(a, S)$ can be evaluated experimentally using the following equation which is obtained by transforming the equation above.

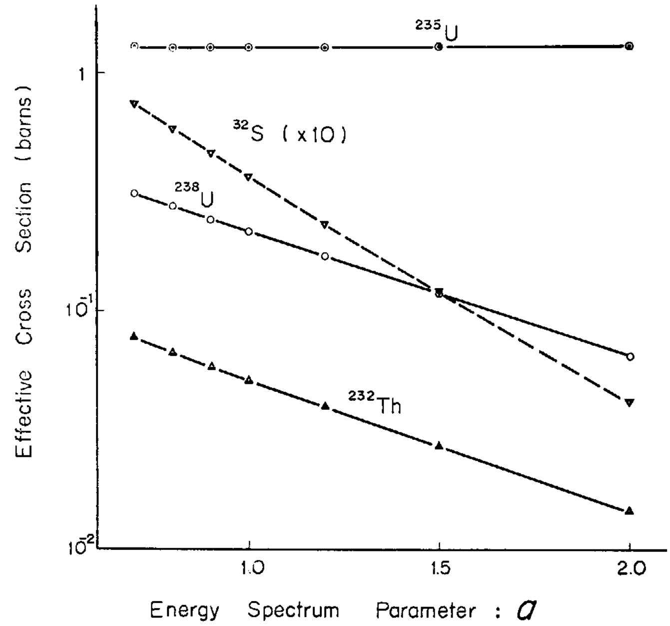

Fig. 5 Effective cross sections of ^{32}S, ^{235}U, ^{238}U, ^{232}Th and ^{237}Np as a function of energy spectrum parameter a.

$$\frac{\sigma_{eff}(a, {}^{232}\mathrm{Th})}{\sigma_{eff}(a, S)} = \frac{\alpha(S)T(a, {}^{232}\mathrm{Th})}{\alpha({}^{232}\mathrm{Th})T(a, S)} . \tag{8}$$

On the other hand, the ratio of $\sigma_{eff}(a, {}^{232}\mathrm{Th})$ to $\sigma_{eff}(a, S)$ as a function of a can be calculated easily. This ratio is shown in **Fig. 6**. The ratios of $\sigma_{eff}(a, {}^{238}\mathrm{U})$ to $\sigma_{eff}(a, S)$ are also shown in **Fig. 6**. If the ratio of $\sigma_{eff}(a, {}^{232}\mathrm{Th})$ to $\sigma_{eff}(a, S)$ is obtained experimentally using the equation (8), the parameter a is determined as an inverse function of the ratio. The values of $\sigma_{eff}(a, {}^{232}\mathrm{Th})$ and $\sigma_{eff}(a, S)$ corresponding to the parameter a are obtained from **Fig. 5**, and thus fast neutron fluence N_f is obtained from equations (5) or (6).

The dose equivalent, DE(a, rem), recommended by ICRP and the surface absorbed dose*), AB57(a, rad), per unit fluence of neutron, whose energy spectrum is given as $\phi_f(a, E)$, are given as follows:

$$\mathrm{DE}(a, \mathrm{rem}) = \int_0^{\infty} \phi_f(a, E)\mathrm{DE}(E)dE \quad , \tag{9}$$

$$\mathrm{AB57}(a, \mathrm{rad}) = \int_0^{\infty} \phi_f(a, E)\mathrm{AB57}(E)dE, \tag{10}$$

---

*) The surface absorbed dose has been taken as the average dose to the front 3 cm thick surface element (No. 57) at the middle of a cylindrical phantom (300 mmϕ × 500 mm) exposed to normally incident neutrons. The surface absorbed dose per unit fluence of neutron was calculated by J. A. Auxier, et al.[7]. This surface absorbed dose is often used in the nuclear accident dosimetry.

where DE(E) and AB57(E) are the dose equivalent and surface absorbed dose per unit fluence of monoenergetic neutron whose energy is E MeV, respectively. DE(a, rem) and AB57(a, rad) as a function of a are shown in **Fig. 7**. For reference, the kerma dose, dose equivalent to the surface

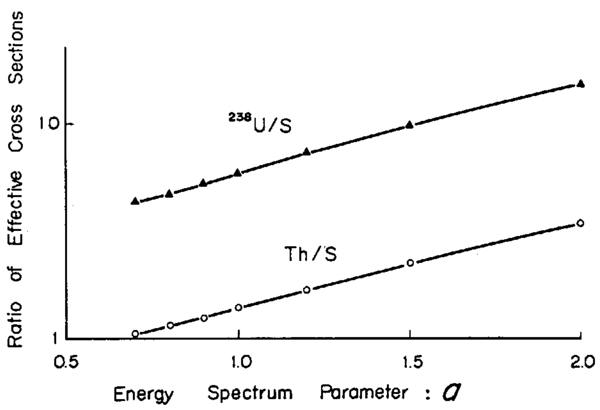

Fig. 6 Ratio of effective cross sections between one atom to another as a function of energy spectrum parameter $\boldsymbol{a}$.

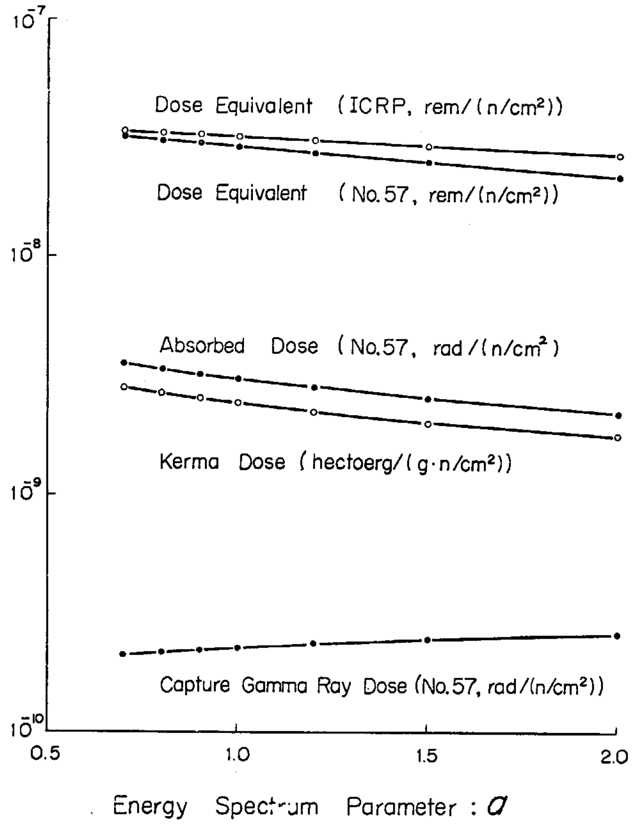

Fig. 7 Dose per neutron unit fluence vs. energy spectrum parameter $\boldsymbol{a}$.

No. 57 represents the front 3 cm thick surface element at the middle of a cylirdrical phantom (300 mmø × 500 mm)[7].

element and capture gamma-ray dose to the surface element per unit fluence of neutron also are shown in **Fig. 7** . As the parameter a is already determined, DE(a, rem) and AB57(a, rad) can be obtained from **Fig. 7** . Finally, dose equivalent or absorbed dose of No. 57 element for fast neutrons of fluence N_f is estimated as the product of N_f and DE(a, rem) or as that of N_f and AB57(a, rad).

4.2 Determination of the parameter n and thermal neutron fluence

Numbers of fission tracks produced by neutron fluences of N_{th}, N_{in} and N_f per unit area of the detectors NU(Cd-Cd), NU(Cd-Sn), NU(Sn-Cd) and EU(Cd-Cd), that is T_1, T_2, T_3 and T_4, respectively, are given as follows:

$$\begin{aligned} T_1 = \alpha(\mathrm{NU}) \, [\, 0.0072 \Big\{ & N_{in} \int_{5\times10^{-7}}^{1} \phi_{in}(n, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE \\ & + N_f \int_0^\infty \phi_f(a, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE \Big\} \\ & + 0.9928 N_f \int_0^\infty \phi_f(a, E)\, \sigma_{^{238}\mathrm{U}}(E)\, dE + RT \,] \,, \end{aligned} \tag{11}$$

$$\begin{aligned} T_2 = \alpha(\mathrm{NU}) \, [\, 0.0072 \Big\{ & N_{in} \int_{5\times10^{-7}}^{1} \phi_{in}(n, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE \\ & + N_f \int_0^\infty \phi_f(a, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE + N_{th} R(\mathrm{th})\, \sigma_{^{235}\mathrm{U}}(\mathrm{th}) \\ & + \sigma_{^{235}\mathrm{U}}(\mathrm{th})\, N_{in} \int_{5\times10^{-7}}^{1} \phi_{in}(n, E)\, R(E)\, dE \\ & + \sigma_{^{235}\mathrm{U}}(\mathrm{th})\, N_f \int_0^\infty \phi_f(a, E)\, R(E)\, dE \Big\} \\ & + 0.9928 N_f \int_0^\infty \phi_f(a, E)\, \sigma_{^{238}\mathrm{U}}(E)\, dE + RT \,] \,, \end{aligned} \tag{12}$$

$$\begin{aligned} T_3 = \alpha(\mathrm{NU}) \, [\, 0.0072 \Big\{ & N_{th}\, \sigma_{^{235}\mathrm{U}}(\mathrm{th}) \\ & + N_{in} \int_{5\times10^{-7}}^{1} \phi_{in}(n, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE \\ & + N_f \int_0^\infty \phi_f(a, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE \Big\} \\ & + 0.9928 N_f \int_0^\infty \phi_f(a, E)\, \sigma_{^{238}\mathrm{U}}(E)\, dE + RT \,] \,, \end{aligned} \tag{13}$$

$$T_4 = \alpha(\mathrm{EU}) \left[0.0144 \left\{ N_{\mathrm{in}} \int_{5\times10^{-7}}^{1} \phi_{\mathrm{in}}(n, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE \right.\right.$$
$$\left. + N_{\mathrm{f}} \int_0^{\infty} \phi_{\mathrm{f}}(a, E)\, \sigma_{^{235}\mathrm{U}}(E)\, dE \right\}$$
$$\left. + 0.9856 N_{\mathrm{f}} \int_0^{\infty} \phi_{\mathrm{f}}(a, E)\, \sigma_{^{238}\mathrm{U}}(E)\, dE + 2RT \right] \quad , \qquad (14)$$

where $\alpha(\mathrm{NU})$ and $\alpha(\mathrm{EU})$ are sensitivities of natural and enriched uranium detectors per unit fluence per unit cross section, respectively, $\sigma_{^{235}\mathrm{U}}(E)$ and $\sigma_{^{238}\mathrm{U}}(E)$ are fission cross sections of $^{235}\mathrm{U}$ and $^{238}\mathrm{U}$ atoms, respectively, $R(E)$ is the fraction of incident neutron fluence reflected as thermal neutrons from a phantom, $\sigma_{^{235}\mathrm{U}}(\mathrm{th})$ is the fission cross section of $^{235}\mathrm{U}$ atoms for thermal neutrons and $R(\mathrm{th})$ is the fraction of incident thermal neutron fluence reflected as thermal neutrons from the phantom. And, $\alpha(\mathrm{NU})RT$ is the number of tracks produced by the neutrons which are reflected and disturbed by the phantom, but having energies greater than 0.5 eV. That is, $\alpha(\mathrm{NU})RT$ is the number of tracks due to the neutrons of which origin is fast or intermediate ($E^n/I(n)$ distribution) neutrons but are not fully thermalized after colliding with the phantom. The value of $\alpha(\mathrm{NU})RT$ is much less than the number of tracks due to the thermal neutrons reflected.

Subtracting equation (11) from equation (13), the following relation is obtained:

$$T_3 - T_1 = 0.0072\, \alpha(\mathrm{NU})\, N_{\mathrm{th}}\, \sigma_{^{235}\mathrm{U}}(\mathrm{th}).$$

As $\alpha(\mathrm{NU})$ can be estimated experimentally, the thermal neutron fluence is obtained as follows:

$$N_{\mathrm{th}} = (T_3 - T_1)/0.0072\, \alpha(\mathrm{NU})\, \sigma_{^{235}\mathrm{U}}(\mathrm{th}) \quad . \qquad (15)$$

Subtracting equation (11) from equation (12), the following equation is obtained:

$$T_2 - T_1 = 0.0072\, \alpha(\mathrm{NU})\sigma_{^{235}\mathrm{U}}(\mathrm{th}) \left\{ N_{\mathrm{th}} R(\mathrm{th}) \right.$$
$$\left. + N_{\mathrm{in}} \int_{5\times10^{-7}}^{1} \phi_{\mathrm{in}}(n, E)\, R(E)\, dE + N_{\mathrm{f}} \int_0^{\infty} \phi_{\mathrm{f}}(a, E)\, R(E)\, dE \right\} . \quad (16)$$

Since $\phi_{\mathrm{f}}(a, E)$ is already evaluated and $R(E)$ is known as illustrated in

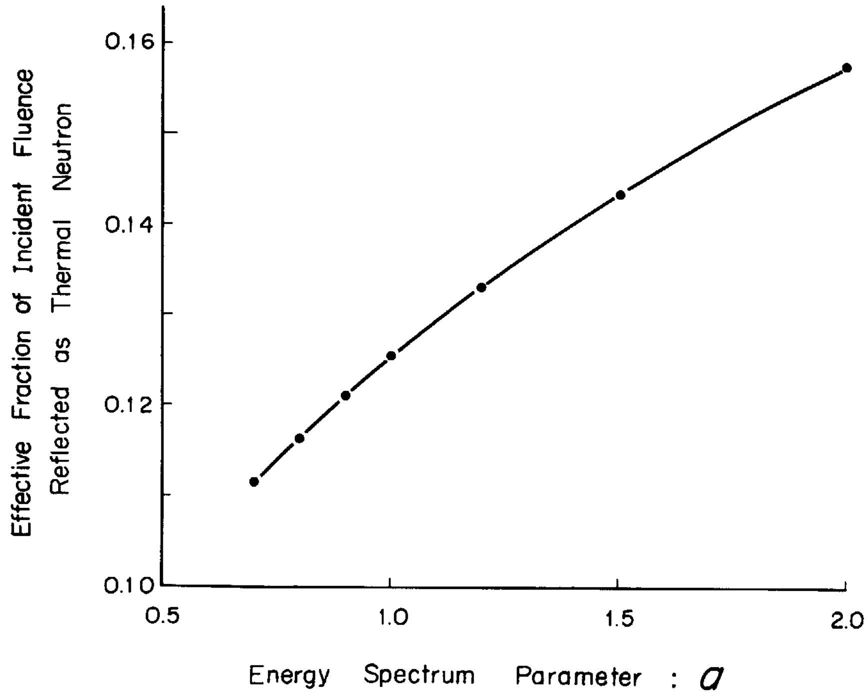

Fig. 8 Effective fraction of incident neutron fluence reflected as thermal neutrons from a phantom vs. energy spectrum parameter a.

Fig. 3, an effective reflection fraction, $R_{eff}(a) \equiv \int_0^\infty \phi_f(a, E)R(E)dE$, can be calculated. The calculated values of $R_{eff}(a)$ are shown in **Fig. 8.** Therefore, the second term in equation (16) can be determined if $T_2 - T_1$ is obtained experimentally. The second term is expressed as follows:

$$0.0072\,\alpha(\mathrm{NU})\sigma_{235_U}(\mathrm{th})N_{in}\int_{5\times10^{-7}}^{1}\phi_{in}(n, E)R(E)dE$$

$$= T_2 - T_1 - 0.0072\,\alpha(\mathrm{NU})\sigma_{235_U}(\mathrm{th})\left\{N_{th}R(\mathrm{th}) + N_f R_{eff}(a)\right\}$$

$$\equiv T(n, R_{eff}) \quad . \tag{17}$$

$T(n, R_{eff})$ represents the number of fission tracks per unit area produced in the NU(Cd-Sn) detector by thermal neutrons, which originated from the incident intermediate neutrons but thermalized and reflected from the body.

Similarly, the first term in equation (11) is expressed as follows:

$$0.0072\,\alpha(\mathrm{NU})N_{in}\int_{5\times10^{-7}}^{1}\phi_{in}(n, E)\,\sigma_{235_U}(E)\,dE$$

$$= T_1 - 0.0072\,\alpha(\mathrm{NU})N_f\sigma_{eff}(a, {}^{235}\mathrm{U})$$

$$- 0.9928\,\alpha(\mathrm{NU})N_f\,\sigma_{\mathrm{eff}}(a,\ {}^{238}\mathrm{U}) - \alpha(\mathrm{NU})RT$$

$$\equiv T(n,\ {}^{235}\mathrm{U})\ , \tag{18}$$

where $\sigma_{\mathrm{eff}}(a,\ {}^{235}\mathrm{U})$ and $\sigma_{\mathrm{eff}}(a,\ {}^{238}\mathrm{U})$ are the effective cross section of ${}^{235}\mathrm{U}$ and ${}^{238}\mathrm{U}$ atoms for $\phi_f(a, E)$, respectively, and they are given in **Fig. 5**. If $\alpha(\mathrm{NU})RT$ is negligible, $T(n,\ {}^{235}\mathrm{U})$ is determined by measuring T_1. $T(n,\ {}^{235}\mathrm{U})$ represents the number of tracks per unit area of the NU(Cd-Cd) detector by directly incident intermediate neutrons ($\phi_{\mathrm{in}} = E^{n}/I\,(n)$) which collide with ${}^{235}\mathrm{U}$ atoms before reaching the phantom.

The ratio of $T(n,\ R_{\mathrm{eff}})$ to $T(n,\ {}^{235}\mathrm{U})$ is expressed as follows:

$$\frac{T(n,\ R_{\mathrm{eff}})}{T(n,\ {}^{235}\mathrm{U})} = \frac{0.0072\alpha(\mathrm{NU})\,\sigma_{^{235}\mathrm{U}}(\mathrm{th})N_{\mathrm{in}}\int_{5\times10^{-7}}^{1}\phi_{\mathrm{in}}(n, E)R(E)\,dE}{0.0072\alpha(\mathrm{NU})N_{\mathrm{in}}\int_{5\times10^{-7}}^{1}\phi_{\mathrm{in}}(n, E)\,\sigma_{^{235}\mathrm{U}}(E)\,dE}$$

$$= \frac{\sigma_{^{235}\mathrm{U}}(\mathrm{th})\int_{5\times10^{-7}}^{1}\phi_{\mathrm{in}}(n, E)R(E)\,dE}{\int_{5\times10^{-7}}^{1}\phi_{\mathrm{in}}(n, E)\,\sigma_{^{235}\mathrm{U}}(E)\,dE}$$

$$= \frac{\sigma_{^{235}\mathrm{U}}(\mathrm{th})R_{\mathrm{eff}}(n)}{\sigma_{\mathrm{eff}}(n,\ {}^{235}\mathrm{U})}\ . \tag{19}$$

Here, it is possible to calculate the effective reflection fraction of intermediate neutrons from the body, i.e., $R_{\mathrm{eff}}(n) \equiv \int_{5\times10^{-7}}^{1}\phi_{\mathrm{in}}(n, E)R(E)\,dE$, and also the effective fission cross section of ${}^{235}\mathrm{U}$ for $\phi_{\mathrm{in}}(n, E)$, i.e., $\sigma_{\mathrm{eff}}(n,\ {}^{235}\mathrm{U}) \equiv \int_{5\times10^{-7}}^{1}\phi_{\mathrm{in}}(n, E)\,\sigma_{^{235}\mathrm{U}}(E)\,dE$. Both results are shown as a function of n in **Fig. 9**. The ratio of $\sigma_{^{235}\mathrm{U}}(\mathrm{th})R_{\mathrm{eff}}(n)$ to $\sigma_{\mathrm{eff}}(n,\ {}^{235}\mathrm{U})$ as a function of n is shown in **Fig. 10**. Since the ratio is determined experimentally as a ratio of $T(n,\ R_{\mathrm{eff}})$ to $T(n,\ {}^{235}\mathrm{U})$, the energy spectrum parameter n is obtained from **Fig. 10** as an inverse function of $\sigma_{^{235}\mathrm{U}}(\mathrm{th})R_{\mathrm{eff}}(n)$ to $\sigma_{\mathrm{eff}}(n,\ {}^{235}\mathrm{U})$ or $T(n,\ R_{\mathrm{eff}})/T(n,\ {}^{235}\mathrm{U})$. If the parameter n is known, the values of $\sigma_{\mathrm{eff}}(n,\ {}^{235}\mathrm{U})$ and $R_{\mathrm{eff}}(n)$ corresponding to the values of n are obtained from **Fig. 9**. Then N_{in} is obtained as follows (see equation (17)):

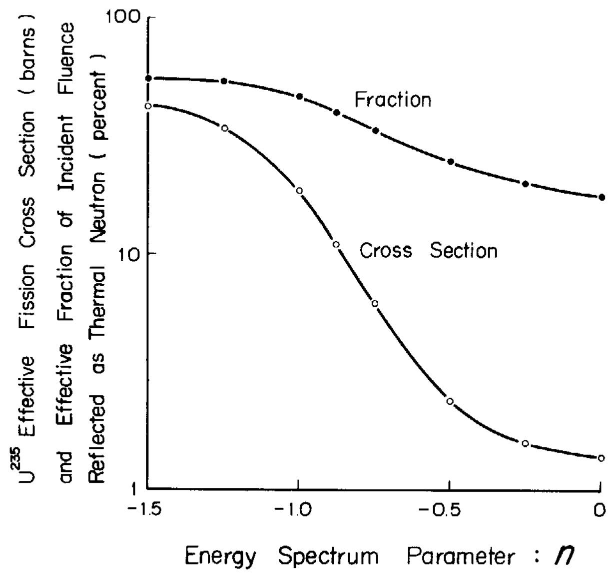

Fig. 9 Effective fission cross section of ^{235}U and effective fraction of incident neutron fluence reflected as thermal neutrons from a phantom vs. energy spectrum parameter n.

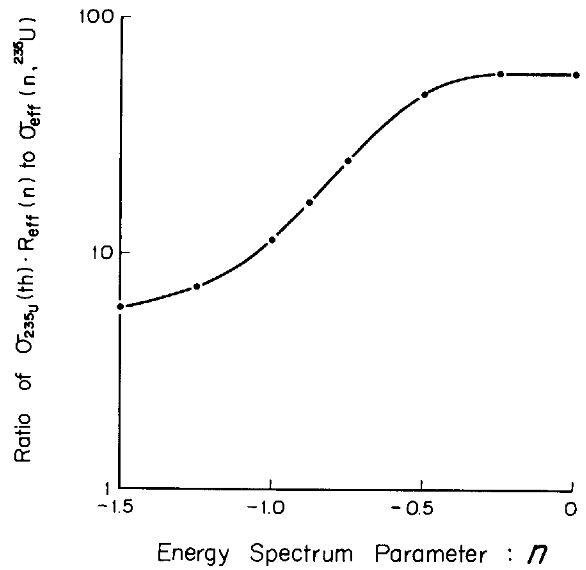

Fig. 10 Ratio of σ_{235U} (th) $\cdot$ $R_{eff}(n)$ to $\sigma_{eff}(n, {}^{235}U)$ vs. energy spectrum parameter n.

$$N_{in} = T(n,\ R_{eff})/0.0072\alpha(\mathrm{NU})\,\sigma_{235_U}(\mathrm{th})\ R_{eff}(n) \tag{20}$$

or (see equation (18)),

$$N_{in} = T(n,\ {}^{235}\mathrm{U})/0.0072\alpha(\mathrm{NU})\,\sigma_{eff}(n,\ {}^{235}\mathrm{U}). \tag{21}$$

The dose equivalent, DE(n, rem), recommended by ICRP and the surface absorbed dose, AB57(n, rad), per unit fluence of intermediate neutrons whose energy spectrum is $\phi_{in}(n, E)$, are given as follows:

$$\mathrm{DE}(n,\ \mathrm{rem}) = \int_{5\times10^{-7}}^{1} \phi_{in}(n, E)\ \mathrm{DE}(E)\,dE, \tag{22}$$

$$\mathrm{AB57}(n,\ \mathrm{rad}) = \int_{5\times10^{-7}}^{1} \phi_{in}(n, E)\,\mathrm{AB57}(E)\,dE. \tag{23}$$

The dose equivalent and absorbed dose as a function of n are shown in **Fig. 11**. For reference, a kerma dose, dose equivalent to the surface element and capture gamma-ray dose to the surface element per unit fluence of intermediate neutron also are shown in **Fig. 11**.

The dose equivalent and the surface absorbed dose for intermediate neutrons of fluence N_{in} are evaluated as the product of N_{in} and DE(n, rem) and that of N_{in} and AB57(n, rad), respectively.

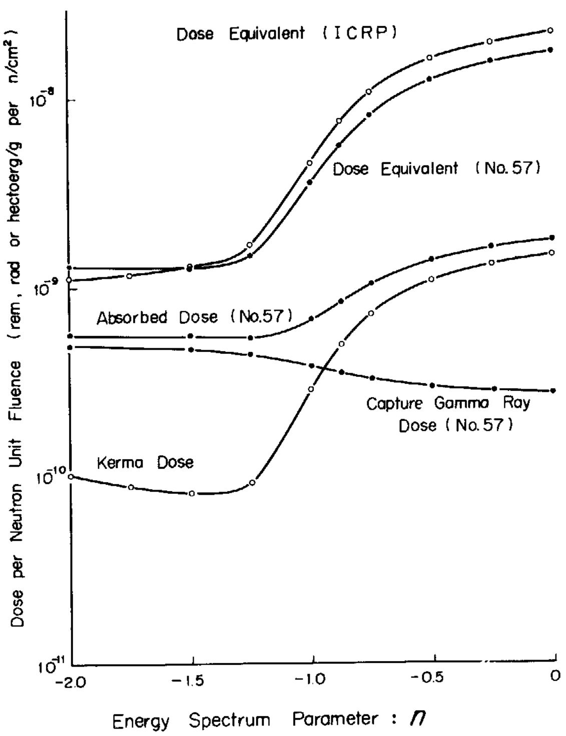

Fig. 11 Dose per unit fluence of neutron vs. energy spectrum parameter n.

No. 57 represents the front 3 cm thick surface element at the middle of a cylindrical phantom (300 mmϕ × 500 mm)[7].

5. Experiment

The present method was applied to neutron dosimetry for a simulated critical accident.

5.1 Detector unit and phantom

The fission track detectors used were ThO_2 or UO_2 doped glass detectors and S-discs. The specification of the detectros is given in **Teble 1**. The content of ^{235}U in EU(Cd-Cd) is 1.45 %, and it is not just twice as much as that in natural uranium, but the difference is small. The arrangement of the detector unit is shown in **Fig. 12**. The reason why two Th-detectors were used was that the cross section of ^{232}Th is too small to get good results with a single detector. A polyethylene bottle (300 mmφ × 350 mm height) filled with water was used as a phantom.

Table 1 Specification of detectors.

Detector	Size (mm)	Weight or Weight percent doped in glass	Enrichment	Sample Number
Sulfur disc	48.5φ x 5.5	20.0 ± 0.1 g		1
ThO_2 doped glass	10 x 10 x 1	28.6 %		2
UO_2 doped glass (natural)	10 x 10 x 1	10.0 %		3
UO_2 doped glass (Enriched)	10 x 10 x 1	9.09%	1.45 %	1

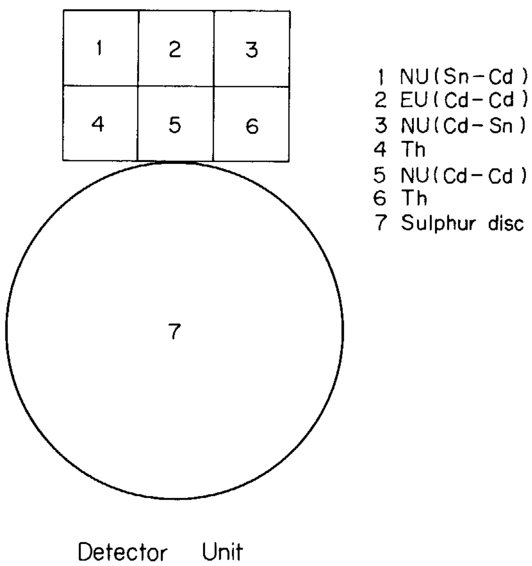

Fig. 12 A detector unit.

5.2 Irradiation facilities of simulated criticality accident

The detector unit was put on the surface of the phantom and was irradiated in the experimental hole of the Japan Research Reactor No. 4 (JRR-4, swimming pool type, 2.5 MW max.). The facilities are shown in **Fig. 13**. When the inner boral shutter was open, the uranium converter was irradiated by neutrons which were slowed down by D_2O. Thus fission neutrons were produced. The specification of the uranium converter is shown in **Table 2**.

The phantom was set on the irradiation wheel at a distance of 60 cm from the wheel head. A moderator, if necessary, was put between the phantom and uranium converter. Moderators consisted of a Lucite plate of 1 or 2 cm in thickness and a pile of paraffin blocks of 90 × 100 cm^2 in area. The total thickness of the moderator was changed from experiment to experiment. If necessary, a cadmium filter was put on the rear side of the moderator. The thickness of the Cd filter is 1 mm and the size is 50 × 100 cm^2.

Irradiations conditions such as power of the reactor, thickness of the moderator, time of irradiation are described in detail in section **6**.

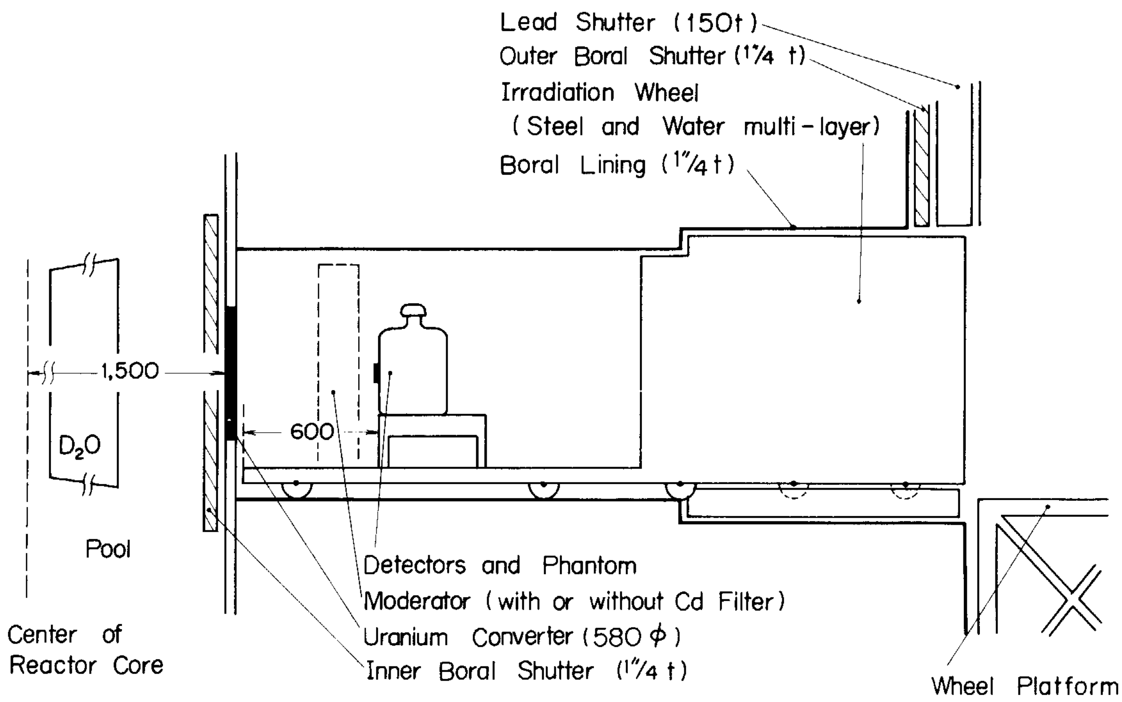

Fig. 13 JRR-4 uranium converter facilities.
(Unit of distance or thickness is millimeter)

Table 2 Specification of uranium converter.

Uranium plate	enrichment	20% metalic
	total weight	15 Kg
	effective dia.	580 mmϕ
	thickness	1.5 mm x 2
Clothing	material	Al
	thickness	2 mm

5.3 Calibration of detectors, etching and counting

A Pu-Be or Am-Be neutron source of 1 Ci was used for the calibration of detectors. The sensitivities of detectors are given as follows.

$$\alpha(\mathrm{S}) = \frac{\text{counts/min}}{\sigma_{\mathrm{eff}}(\mathrm{S},\ \text{Pu-Be})\cdot\text{fluence}} \quad \text{or} \quad \alpha(\mathrm{S}) = \frac{\text{counts/min}}{\sigma_{\mathrm{eff}}(\mathrm{S},\ \text{Am-Be})\cdot\text{fluence}}$$

$$\alpha(\mathrm{Th}) = \frac{\text{tracks/cm}^2}{\sigma_{\mathrm{eff}}(\mathrm{Th},\text{Pu-Be})\cdot\text{fluence}} \quad \text{or} \quad \alpha(\mathrm{Th}) = \frac{\text{tracks/cm}^2}{\sigma_{\mathrm{eff}}(\mathrm{Th},\text{Am-Be})\cdot\text{fluence}},$$

$$\alpha(\mathrm{NU}) = \frac{\text{tracks/cm}^2}{\sigma_{\mathrm{eff}}(\mathrm{NU},\text{Pu-Be})\cdot\text{fluence}} \quad \text{or} \quad \alpha(\mathrm{NU}) = \frac{\text{tracks/cm}^2}{\sigma_{\mathrm{eff}}(\mathrm{NU},\text{Am-Be})\cdot\text{fluence}},$$

and

$$\alpha(\mathrm{EU}) = \frac{\text{tracks/cm}^2}{\sigma_{\mathrm{eff}}(\mathrm{EU},\text{Pu-Be})\cdot\text{fluence}} \quad \text{or} \quad \alpha(\mathrm{EU}) = \frac{\text{tracks/cm}^2}{\sigma_{\mathrm{eff}}(\mathrm{EU},\text{Am-Be})\cdot\text{fluence}},$$

where $\sigma_{\mathrm{eff}}(\mathrm{S},\ \text{Pu-Be})$, $\sigma_{\mathrm{eff}}(\mathrm{S},\ \text{Am-Be})$, ..., $\sigma_{\mathrm{eff}}(\mathrm{EU},\ \text{Pu-Be})$ and $\sigma_{\mathrm{eff}}(\mathrm{EU},\ \text{Am-Be})$ are effective cross sections of S, thorium, natural uranium and enriched uranium for the Pu-Be or Am-Be neutron source, respectively.

The fission track detectors were etched in 20 weight percent NaOH solution at a temperature of 50 ± 2°C. The etching time was 10 minutes for ThO_2 doped glass plates and 20 minutes for UO_2 doped glass plates. The counting of tracks was made with a microscope of 562.5 magnifications.

The activities of the S-discs were measured with a 2 in.ϕ GM counter directly without treatment such as burning.

6. Results and Discussion

The first experiment was performed four times from Feb. 24 to Mar. 4, 1971. The irradiation conditions were 1) 200 kW, 20 minutes irradiation without any moderator and 2) 200 kW, 30 minutes irradiation with a 2 cm Lucite moderator. The experimental results on responses (cpm and tracks/cm^2) and the sensitivities of the detectors are given in **Table 3**. The values of Th detector are the averages of two detectors. The fluence and the dose

Table 3 Responses and sensitivities of detectors, (Reb. 24 ~ Mar. 4, 1971).

Response (cpm or tracks/cm^2)

Condition	200kW, 20 min., no moderator		20kW, 30 min. 2cm Lucite	
	Feb. 24	Feb. 25	Mar. 3	Mar. 4
S - disc	219	228	259	250
Th	456	482	543	511
NU(Sn-Cd)	8.20 x 10^3	7.80 x 10^3	10.5 x 10^3	11.7 x 10^3
NU(Cd-Sn)	11.3 x 10^3	12.6 x 10^3	12.9 x 10^3	14.3 x 10^3
NU(Cd-Cd)	2.32 x 10^3	2.24 x 10^3	2.62x 10^3	2.92x 10^3
EU(Cd-Cd)	3.23 x 10^3	3.10 x 10^3	3.68x 10^3	3.78x 10^3

Sensitivity

α(S)	4.98 x 10^{-7} cpm/(fluence·barn)
α(Th)	8.27 x 10^{-7} (track/cm^2)/(fluence·barn)
α(NU)	4.88 x 10^{-7} (track/cm^2)/(fluence·barn)
α(EU)	4.40 x 10^{-7} (track/cm^2)/(fluence·barn)

were evaluated by the method described in section **4.** For example, the energy spectrum parameter a was obtained from equation (8) and **Fig. 6.** That is, first, the ratio of $\sigma_{eff}(a, {}^{232}Th)$ to $\sigma_{eff}(a, S)$ was determined on the condition without moderator as follows:

$$\frac{\sigma_{eff}(a, {}^{232}Th)}{\sigma_{eff}(a, S)} = \frac{\alpha(S)T(a, {}^{232}Th)}{\alpha({}^{232}Th)T(a, S)} = \frac{4.98 \times 10^{-7}}{8.27 \times 10^{-7}} \times \frac{456}{219} = 1.25 ,$$

and then the parameter a was determined as 0.89 from **Fig. 6**. The fluence and the dose evaluated are given in **Table 4**. Dose-1, dose-2, dose-3, dose-4 and dose-5 in **Table 4.** shows the dose equivalent recommended by ICRP,

Table 4 Fluence and dose assessment, (Feb. 24 ~ Mar. 4, 1971).

	Date	Condition	Fast Neutron	Thermal Neutron
Energy Spectrum Parameter	Feb. 24	1 200kW, 20min. no moderator	0.89	
	25		0.90	
	Mar. 3	2 200kW, 30min. 2cm Lucite	0.90	
	4		0.87	
Fluence (n/cm^2)	Feb. 24	1	9.31×10^9	2.89×10^9
	25		9.93×10^9	2.73×10^9
	Mar. 3	2	11.2×10^9	3.86×10^9
	4		10.2×10^9	4.32×10^9
Dose 1 (rem)	Feb. 24	1	297	3
	25		316	3
	Mar. 3	2	356	4
	4		326	5
Dose 2 (hectoerg/g)	Feb. 24	1	24.8	0.1
	25		25.1	0.1
	Mar. 3	2	28.4	0.1
	4		26.2	0.1
Dose 3 (rem)	Feb. 24	1	279	3
	25		298	3
	Mar. 33	2	336	4
	4		308	5
Dose 4 (rad)	Feb. 24	1	29.8	1.4
	25		31.9	1.3
	Mar. 3	2	36.0	0.18
	4		33.2	0.20
Dose 5 (rad)	Feb. 24	1	2.05	1.16
	25		2.20	1.09
	Mar. 3	2	2.49	1.54
	4		2.24	1.73

Dose-1, dose-2, dose-3, dose-4 and dose-5 in Table 4 shows the dose equivalent recommended by ICRP, kerma dose, dose equivalent to the surface element, surface absorbed dose and capture gamma-ray dose to the surface element, respectively.

kerma dose, dose equivalent to the surface element, surface absorbed dose and capture gamma-ray dose to the surface element, respectively. The parameter a obtained in the experiment with the moderator of 2 cm Lucite was almost equal to that obtained without moderator. The moderator of 2 cm Lucite did not affect on the shape of the fast neutron spectrum. As the ratio of $T(n, R_{eff})$ to $T(n, {}^{235}U)$ was very small (2 ~ 4) the parameter n could not be obtained. (The ratio must be larger than 5.9 as shown in **Fig. 10**, although $\alpha(NU)RT$ is ignored).

In the second experiment, a thick moderator composed of a 15 cm paraffin pile and a 3 cm Lucite plate was used to change the energy spectrum of neutrons. The power of the reactor was 2.5 MW and the irradiation time was 30 minutes. The responses of the detectors and the sensitivities are given in **Table 5.** The results of fluence and dose assessment are shown in **Table 6**. The value of the parameter a of 1.05 was larger than 0.89 which was obtained without moderator and the average energy of the fast neutrons was lower than that in the first experiment. Again, the parameter n for intermediate neutrons could not be evaluated.

Table 5 Responses and sensitivities of detectors, (May 11, 1971).

Response (cpm or tracks/cm^2)

Condition	2.5 MW, 30 min., 15 cm paraffin plus 3 cm Lucite		
	No. 1	No. 2	No. 3
S - disc	367	365	364
Th	841	847	853
NU(Sn-Cd)	22.3×10^3	22.1×10^3	22.2×10^3
NU(Cd-Sn)	17.0×10^3	13.7×10^3	14.5×10^3
NU(Cd-Cd)	2.85×10^3	2.68×10^3	2.70×10^3
EU(Cd-Cd)	3.38×10^3	3.60×10^3	3.28×10^3

Sensitivity

α(S)	5.15×10^{-7} cpm/(fluence·barn)
α(Th)	8.15×10^{-7} (tracks/cm^2)/(fluence·barn)
α(NU)	5.10×10^{-7} (tracks/cm^2)/(fluence·barn)
α(EU)	4.55×10^{-7} (tracks/cm^2)/(fluence·barn)

Table 6 Fluence and dose assessment, (May 11, 1971).

	Condition	Fast Neutron	Thermal Neutron
Energy Spectrum Parameter	2.5 MW, 30 min., 15 cm paraffin & 3 cm Lucite	1.04 1.05 1.06	
Fluence (n/cm^2)	"	2.12×10^{10} 2.17×10^{10} 2.21×10^{10}	9.18×10^9 9.13×10^9 9.20×10^9
Dose 1 (rem)	"	651 664 675	10 9 10
Dose 2 (hectoerg/g)	"	50.4 51.5 52.2	0.2 0.2 0.2
Dose 3 (rem)	"	609 619 630	11 11 11
Dose 4 (rad)	"	63.7 65.1 66.1	4.3 4.3 4.3
Dose 5 (rad)	"	4.81 4.95 5.04	3.77 3.75 3.78

Dose-1, dose-2, dose-3, dose-4 and dose-5 in Table 4 shows the dose equivalent recommended by ICRP, kerma dose, dose equivalent to the surface element, surface absorbed dose and capture gamma-ray dose to the surface element, respectively.

In the third experiment, the rear side of the moderator (3 cm Lucite + 15 cm paraffin) was covered with a cadmium filter whose thickness was 1 mm for cutting off thermal neutrons. The size of the cadmium filter was 50 cm in height and 100 cm in width. The experimental results are given in **Tables 7** and **8**. The tables include also the results obtained with a moderator of 1 cm Lucite plus Cd filter. Thermal neutrons were not cut off completely. Therefore, in the fourth experiment a large cadmium filter was used. The size was increased to 90 cm in height and 100 cm in width.

The results are given in **Tables 9** and **10.** Thermal neutron fluence decreased, but the effect of cadmium filter on cutting off thermal neutrons was not complete. The parameter n for intermediate neutrons and subsequently the fluence for intermediate neutrons, N_{in}, could not be evaluated again in the third and fourth experiments.

Here it may be appropriate to discuss the reasons why the ratio of $T(n, R_{eff})$ to $T(n, {}^{235}U)$ obtained were too small to estimate the parameter n under the present experimental conditions.

In the first to fourth experiments, the $T(n, {}^{235}U)$ was simply estimated as T_1 minus the contribution of fast neutrons, assuming that the contribution of $\alpha(NU)RT$ to the T_1(tracks/cm^2 in the detector) was small.

This assumption may introduce an error in the estimation of $T(n, {}^{235}U)$. If the contribution of $\alpha(NU)RT$ was fairly large as compared with the total tracks minus the tracks due to fast neutrons, the disregard of the $\alpha(NU)RT$ resulted in a large $T(n, {}^{235}U)$ value than the true one. Thus, the ratio of $T(n, R_{eff})$ to $T(n, {}^{235}U)$ must be underestimated.

Table 7 Responses and sensitivities of detectors, (Aug. 27 ~ Sept. 3, 1971).

Response (cpm or tracks/cm^2)

Condition	2.5 MW, 30 min., 18 cm mod.+ Cd filter			200 kW, 20 min., 1 cm 1 cm Lucite + Cd filter		
Date	Aug.27, No.1	Aug.27, No.2	Aug.27, No.3	Sep. 1	Sep. 2	Sep. 3
S	227	237	232	204	194	194
Th	787	781	779	630	560	586
NU(Sn-Cd)	3.63×10^3	4.13×10^3	3.89×10^3	4.40×10^3	4.71×10^3	4.55×10^3
NU(Cd-Sn)	6.97×10^3	7.59×10^3	7.19×10^3	13.9×10^3	12.9×10^3	12.8×10^3
NU(Cd-Cd)	2.35×10^3	2.47×10^3	2.38×10^3	3.04×10^3	2.95×10^3	3.27×10^3
EU(Cd-Cd)	2.97×10^3	3.20×10^3	3.00×10^3	4.67×10^3	4.46×10^3	4.34×10^3

Sensitivity

α(S)	3.06×10^{-7} cpm/(fluence barn)
α(Th)	7.43×10^{-7} (tracks/cm^2)/(fluence·barn)
α(NU)	4.46×10^{-7} (tracks/cm^2)/(fluence·barn)
α(EU)	4.00×10^{-7} (tracks/cm^2)/(fluence·barn)

Table 8 Fluence and dose assessment, (Aug. 27 ~ Sept. 3, 1971).

	Date	Condition	Fast Neutron	Thermal Neutron
Energy Spectrum Parameter	Aug.27	1 2.5MW, 30min. 18 cm mod.+ Cd filter	1.03 0.98 0.99	
	Sep. 1 2 3	2 200kW, 30min. 1cm Lucite + Cd filter	0.91 0.84 0.88	
Fluence (n/cm^2)	Aug.27	1	2.16×10^{10} 2.01×10^{10} 2.02×10^{10}	6.92×10^{8} 8.96×10^{8} 8.16×10^{8}
	Sep. 1 2 3	2	1.48×10^{10} 1.20×10^{10} 1.32×10^{10}	7.35×10^{8} 9.52×10^{8} 6.92×10^{8}
Dose 1 (rem)	Aug.27	1	665 625 626	1 1 1
	Sep. 1 2 3	2	471 386 421	1 1 1
Dose 2 (hectoerg/g)	Aug.27	1	51.6 49.1 48.9	- - -
	Sep. 1 2 3	2	37.2 31.3 33.8	- - -
Dose 3 (rem)	Aug.27	1	622 585 586	1 1 1
	Sep. 1 2 3	2	442 365 396	1 1 1
Dose 4 (rad)	Aug.27	1	65.2 62.1 62.2	0.3 0.4 0.4
	Sep. 1 2 3	2	47.3 39.6 42.8	0.3 0.4 0.3
Dose 5 (rad)	Aug.27	1	4.90 4.52 4.57	0.28 0.36 0.33
	Sep. 1 2 3	2	3.14 2.63 2.90	0.29 0.38 0.28

Dose-1, dose-2, dose-3, dose-4 and dose-5 in Table 4 shows the dose equivalent recommended by ICRP, kerma dose, dose equivalent to the surface element, surface absorbed dose and capture gamma-ray dose to the surface element, respectively.

Table 9 Responses and sensitivities of detectors, (May 18 ~ 26, 1972).

Response (cpm or tracks/cm^2)

Condition	200 kW, 25 min., 1 cm Lucite		200 kW, 25 min., 1 cm Lucite + Cd filter	
Date	May 18	May 19	May 25	May 26
S	271	252	253	247
Th	553	510	495	505
NU (Sn-Cd)	12.6×10^3	11.6×10^3	2.86×10^3	2.98×10^3
NU (CD-Sn)	15.6×10^3	14.1×10^3	8.95×10^3	9.22×10^3
NU (Cd-Cd)	2.75×10^3	2.73×10^3	2.37×10^3	2.53×10^3
EU (Cd-Cd)	3.88×10^3	3.68×10^3	3.67×10^3	3.76×10^3

Sensitivity

α(S)	5.26×10^{-7} cpm/(fluence·barn)
α(Th)	8.32×10^{-7} (tracks/cm^2)/(fluence·barn)
α(NU)	5.27×10^{-7} (tracks/cm^2)/(fluence.barn)
α(EU)	4.81×10^{-7} (tracks/cm^2)/(fluence·barn)

Table 10 Fluence and dose assessment, (May 18 ~ 26, 1972).

	Date	Condition	Fast Neutron	Thermal Neutron
Energy Spectrum Parameter	May 18 19	1 200 kW, 25 min., 1 cm Lucite	0.92 0.91	
	May 25 26	2 200kW, 25min., 1cm Lucite + Cd filter	0.88 0.92	
Fluence (n/cm^2)	May 18 19	1	1.17×10^{10} 1.07×10^{10}	4.49×10^9 4.07×10^9
	May 25 26	2	1.00×10^{10} 1.07×10^{10}	0.22×10^9 0.20×10^9
Dose 1 (rem)	May 18 19	1	370 339	5 4
	May 25 26	2	319 338	- -
Dose 2 (hectoerg/g)	May 18 19	1	29.3 27.0	0.1 0.1
	May 25 26	2	25.5 26.8	- -
Dose 3 (rem)	May 18 19	1	349 320	5 5
	May 25 26	2	299 319	- -
Dose 4 (rad)	May 18 19	1	37.2 34.2	2.1 1.9
	May 25 26	2	32.2 34.0	0.1 0.1
Dose 5 (rad)	May 18 19	1	2.61 2.37	1.80 1.63
	May 25 26	2	2.20 2.39	0.09 0.08

Dose-1, dose-2, dose-3, dose-4 and dose-5 in Table 4 shows the dose equivalent recommended by ICRP, kerma dose, dose equivalent to the surface element, surface absorbed dose and capture gamma-ray dose to the surface element, respectively.

Making sure it, the $\alpha(NU)RT$ was estimated by the fifth experiments. First, the detector unit was set on a thin aluminum plate without phantom and any moderator. The position of the detector unit (*i. e.*, the height over the wheel deck and the distance from the wheel head) was the same as that of the detector unit on the phantom, and the power of the reactor was 200 kW and irradiation time was for 25 minutes. Second, the detector unit was set on the phantom without moderator (200 kW, 25 min.). Third, a Cd filter of 90 × 100 cm^2 in area was placed between the uranium converter and the phantom (200 kW, 25 min.). Last, a moderator of 5 cm paraffin pile and 3 cm Lucite plate was placed in front of the phantom (200 kW, 1 hour). The responses and the sensitivities of the detectors are given in **Table 11**. **Table 12** shows the fast and thermal neutron fluences on these conditions. Counting rates of the S-discs shown in the three columns in the middle of **Table 11** are closely equal to each other and the numbers of tracks/cm^2 of the Th-detectors in the same columns are also closely equal to each other. It shows that the disturbance of neutron energy spectrum by the phantom did not occur in the energy range greater than 2 MeV. The difference between the number of tracks/cm^2 of NU(Cd-Cd) in the column of with phantom or phantom + Cd filter and that in the column of without phantom is about 500 tracks/cm^2. This difference corresponds to α(NU)RT. If the tracks, 500 tracks/cm^2, which were produced by the neutrons being reflected from the phantom and having the energies greater than 0.5 eV (the origin of the neutrons was the fast or intermediate ($\phi_n = E^n/I(n)$) neutrons), are ignored, the ratio of $T(n, R_{eff})$ to $T(n, {}^{235}U)$ becomes 3.3, but if α(NU)RT = 500 tracks/cm^2 is taken into account the ratio becomes 4.8. However, the disregard of the α(NU)RT may not explain enough the smaller values of the ratio.

Table 11 Responses and sensitivities of detectors (Sept. 6 ~ 29, 1972).

Response (cpm or tracks/cm^2)

Condition	200 kW, 25 min.			200 kW, 1 hour with phantom 5cm paraffin + 3 cm Lucite
	Without phantom	With phantom	Phantom + Cd filter	
S	293	286	295	219
Th	613	600	621	456
NU(Sn-Cd)	10.5 x 10^3	11.2 x 10^3	3.26 x 10^3	20.8 x 10^3
NU(Cd-Sn)	3.05 x 10^3	16.1 x 10^3	12.1 x 10^3	13.2 x 10^3
NU(Cd-Cd)	2.63 x 10^3	3.13 x 10^3	3.12 x 10^3	1.82 x 10^3
EU(Cd-Cd)	3.31 x 10^3	3.92 x 10^3	4.09 x 10^3	2.52 x 10^3

Sensitivity

α(S)	5.10 x 10^{-7} cpm/(fluence·barn)
α(Th)	8.23 x 10^{-7} (tracks/cm^2)/(fluence·barn)
α(NU)	5.04 x 10^{-7} (tracks/cm^2)/(fluence·barn)
α(EU)	4.58 x 10^{-7} (tracks/cm^2)/(fluence·barn)

Table 12 Fluence and dose assessment, (Sept. 6 ~ 29, 1972).

	Irradiation Condition	Fast Neutron
Energy Spectrum Parameter	Without Phantom	0.92
	Phantom	0.93
	Phantom + Cd Filter	0.93
	Phantom + Moderator	0.92
Fluence	Without Phantom	1.31×10^{10}
	Phantom	1.29×10^{10}
	Phantom + Cd Filter	1.35×10^{10}
	Phantom + Moderator	0.98×10^{10}
Dose 1 (rem)	Without Phatom	414
	Phantom	406
	Phantom + Cd Filter	425
	Phantom + Moderator	309
Dose 2 (hectoerg/g)	Without Phatom	32.8
	Phantom	32.2
	Phantom + Cd Filter	33.7
	Phantom + Moderator	24.5
Dose 3 (rem)	Without Phantom	390
	Phantom	383
	Phantom + Cd Filter	400
	Phantom + Moderator	292
Dose 4 (rad)	Without Phantom	41.5
	Phantom	40.6
	Phantom + Cd Filter	42.5
	Phantom + Moderator	31.0
Dose 5 (rad)	Without Phantom	2.91
	Phantom	2.88
	Phantom + Cd Filter	3.01
	Phantom + Moderator	2.17

Dose-1, dose-2, dose-3, dose-4 and dose-5 in Table 4 shows the dose equivalent recommended by ICRP, kerma dose, dose equivalent to the surface element, surface absorbed dose and capture gamma-ray dose to the surface element, respectively.

Accordingly, another reason was looked for. It was suspected that low energy neutrons having the energies of eV order other than thoses represented by $E^n/I(n)$ distribution might exist predominantly in the incident neutrons under the present experimental conditions. If the fluence of intermediate neutrons which are expressed by $E^n/I(n)$ distribution was N_1 and that of eV order neutrons was N_2, the ratio of tracks produced by these total neutrons after being thermalized and reflected by the phantom to those due to the direct incident neutrons is expressed as follows:

$$\frac{N_1\sigma(\mathrm{th}) \times R(E^n) + N_2\sigma(\mathrm{th}) \times R(\mathrm{eV})}{N_1\sigma(E^n) + N_2\sigma(\mathrm{eV})} ,$$

here $R(\mathrm{eV})$ represents the fraction of the eV order neutrons reflected as thermal neutrons from the phantom, $\sigma(E^n)$ is the average cross section of ^{235}U for the intermediate neutrons whose energies are expressed by E^n distribution and $\sigma(\mathrm{eV})$ is the cross section of neutrons of eV order. The value of $\sigma(E^n)$ is unknown. If $n=-1$, $\sigma(E^{-1}) = 18.6$ barns. Assuming $\sigma(\mathrm{eV})$ is 100 barns and taking $R(E^{-1})$ and $R(\mathrm{eV})$ are 0.46 and 0.55, respectively, the above ratio is expressed in the following way:

$$\frac{577(N_1 \times 0.46 + N_2 \times 0.55)}{N_1 \times 18.6 + N_2 \times 100} .$$

Since the ratio of N_1 to N_2 is unknown, the above ratio can not be decided. If $N_1 \ll N_2$, the ratio becomes about 3. If $N_1 = N_2$, the ratio becomes about 4.9. As the experimental values of the ratio was 4.8 after correction of $\alpha(\mathrm{NU})RT$, it seems that $N_1 = N_2$. Therefore, it may be concluded that the intermediate neutrons ($0.5 \times 10^{-6} \sim 1$ MeV) were not represented by $E^n/I(n)$ distribution under the present experimental conditions, and the neutrons of eV order exist considerablly in the incident intermediate neutrons.

The presence of low energy neutrons (eV order) was demonstrated by means of gold foils (255 mg/cm^2) placing on an aluminum plate when neither the phantom nor moderator was used. The foils were irradiated for 20 minutes at 200 kW. The counting rate of the induced activity of the front side of the gold foils of which both sides were covered with the cadmium filter was 2.4 times as large as that of the rear side. It means that the activation cross section was very large as well known and the incident neutron were absorbed profoundly in the gold foils. The largest cross section of gold is observed at 4.9 eV where resonance occurs. Thus, the presence of high flux density of low energy neutrons (eV order) in the irradiation field was detected. The cadmium ratio of the neutrons moderated by D_2O and light water in the pool of JRR-4 (see **Fig. 13**) is small (about 20, Au foils used*). It was likely that low energy neutrons from the reactor passed through the uranium converter and were detected with NU(Cd-Cd), NU(Cd-Sn), NU(Sn-Cd) and EU(Cd-Cd).

As shown in **Table 12**, the doses due to the intermediate neutrons could not be evaluated because the fluence of these neutrons were unknown. As described above, it was suspected that the fluence of neutrons in the resonance range was comparable to that of the intermediated neutrons of E^n-distribution. If the effective cross sections of ^{235}U atoms for neutrons in the resonance range and for neutrons of E^n-distribution were 100 and 18.6 barns, respectively, these two fluences were the same, and the total fluence of the intermediate energies becomes 4.8×10^9 n/cm^2. This was slightly larger than the thermal fluence, 3.8×10^9 n/cm^2. As the dose equivalent (ICRP) per unit fluence of neutron whose energy is a few eV is about 10^{-9} rem/n/cm^2, the dose due to the total intermediate neutrons including resonance neutrons was perhaps less than 10 rem. In this case, the contribution of the intermediate neutrons to the total dose is only a few percents which is less than the experimental error in the estimation of fast neutron dose.

For comparison of all results, fluences are normalized to 2.5 MW and one hour irradiation. They are given in **Table 13**. When the thickness of the moderator was less than 8 cm, the parameter a was about 0.90 and was equal to the value obtained without moderator. The average energy of the fast neutrons produced in the uranium converter facilities without moderater in the JRR-4 was estimated to be 1.67 MeV ($= 3/(2a)$ MeV). This is obviously lower than 2 MeV which was obtained as the average energy for the uncollided fission neutrons ($a = 0.776$). The moderation of neutrons by the aluminium clothing of metalic uranium and by the pool wall and the scattering of neutrons by the wall of the irradiation hole and by the irradiation wheel may explain the lower average energy. The energy spectrum parameter a of the neutrons moderated by 15 cm paraffin plus 3 cm Lucite was estimated 1.00 ~ 1.05 as shown in the last two lines in **Table 12**, and then the average

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* Private communication from Mr. Morozumi, a member of the JRR-4 Operation Section.

was 1.4 ~ 1.5 MeV and was lower than that obtained without moderation, 1.67 MeV. The comparison between the spectrum obtained without moderator and that obtained with moderator of 15 cm paraffin plus 3 cm Lucite is shown in **Fig. 14**. Thus, the change in energy spectrum is given as the change of energy spectrum parameter a. This is the merits of the new accident dosimetry system in JAERI.

Dosage detection limit of this system under the present experimental conditions (energy spectrum, etching, counting and so on) is about 50 rems as ICRP dose equivalent or 10 rads as surface absorbed dose.

Table 13 Fluence normalized to 1 hour irradiation at 2.5 MW.

Condition			Energy Spectrum Parameter	Neutron Fluence($\times 10^{10}$m/cm^2)	
Phantom	Moderator	Cd Filter	a	Fast	Thermal
NO	NO	NO	0.92	39.3	11.2
Yes	NO	NO	0.90	36.1	10.6
			0.93	38.7	11.5
		Yes*	0.93	40.5	-
Y Yes	Lucite 1cm	NO	0.91	33.6	12.8
		Yes**	0.88	33.3	1.98
		Yes*	0.90	31.0	0.63
Yes	Lucite 2 cm	NO	0.89	26.6	10.2
Yes	Lucite 3cm + Paraffin 5cm	NO	0.92	12.2	9.3
Yes	Lucite 3cm + Paraffin 15cm	NO	1.05	4.33	1.82
		Yes**	1.00	4.13	0.16

* Cd Filter Size 900 x 1000 x 1 (mm)
** Cd Filter Size 500 x 1000 x 1 (mm)
Lucite, Paraffin 900 x 1000 (mm)

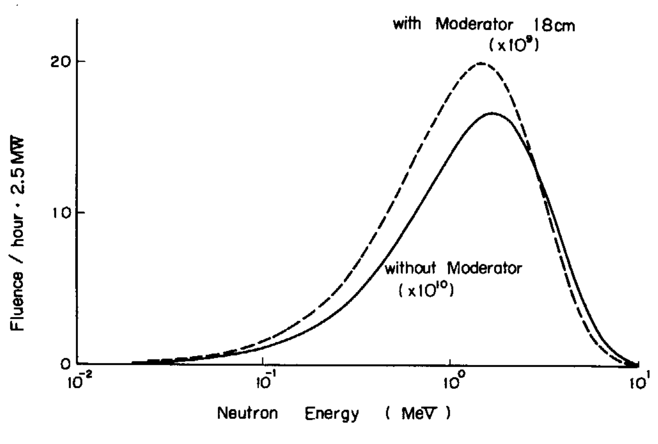

Fig. 14 Energy spectra in JRR-4 uranium converter facilities.

Acknowledgement

The authors would like to express their thanks to Mr. K. Kawai and Mr. T. Watanabe for their assistance with experiments, to the members of JRR-4 Operation Section, to Dr. K. Tsukada, Mr. C. Kobayashi and Mr. S. Kanda for use of the Van de Graaff accelerator used for calibration of detectors and to Dr. K. Nishimura for guidances in the evaluation of neutron cross sections.

They are also grateful to Dr. I. Miyanaga and Dr. H. Tatsuta for their encouragements. Last but not least, they are gratefule to the members of the Radiation Dosimetry Section for their discussions.

References

1) Saito, C. *et al.* ; Dosimetry in Nuclear Accidents, JAERI-memo 3323 (1968), (in Japanese).
2) Katsuragi, S. *et al.* ; JAERI 1104 (1956), (in Japanese).
3) Dennis, J. A. ; JAERI-memo 1967 (1965).
4) Cranberg, L. *et al.* ; *Phys. Rev.* **103**, 662 (1956).
5) Yokota, R. *et al.* ; *Nucl. Instr. Meth.* **61**, 119 (1968).
6) Tatsuta, H. *et al.* ; *J. Atomic Energy Soc. Japan* **9**, 700 (1967), (in Japanese).
7) Auxier, J. A. *et al.* ; "Radiation Dosimetry" edited by Attix, F. H. and Roesch, W. C., Vol. 1, (1968).