*Journal of the Korean Nuclear Society*
*Vol. 2, No.2, June, 1970*

# 《Original》 Fast Neutron Dosimetry with Two Threshold Detectors in Criticality Accidents of Nuclear Reactors

Seung Gy Ro

Reactor Centrum Nederland, Petten, The Netherlands

(Received May 6, 1970)

**Abstract**

An attempt has been made to do interpretation of the fast neutron dose with two threshold detectors incorporated with the Harwell criticality locket. This method is based on the assumption that the spectral distribution of fission neutrons in criticality accidents may be governed by one spectral parameter.

The surface-absorbed dose for a unit fission neutron fluence seems to be insensitive to spectral shifts of the fission neutron spectrum. The average cross-sections for the activation detectors, however, are considerably changed with the neutron spectral shape, which may lead to a large error in calculating the dose from the reaction rate if one uses a fixed value for the average cross sections regardless of the neutron spectral distribution.

Besides, the doses calculated from three representative formulae for fission neutron spectra have been compared: these formulae are Watt, Cranberg at al. and Maxwellian forms. The results obtained from the Maxwellian formula show a departure from the Watt and Cranberg's, both being similarly close.

**요 약**

두개의 threshold detector로서 원자로의 폭발사고시에 방출되는 속 중성자의 속도분포를 측정하고 그로부터 속 중성자의 인체흡수선량을 계산하였다. 이때 속 중성자의 속도분포는 하나의 스펙트럼 매개변수에 의하여 결정된다는 가정으로부터 얻어지는데 이 매개변수는 threshold detector의 반응율을 측정함으로서 구해진다.

속 중성자의 인체흡수선량은 속 중성자의 속도분포 변화에 따라 큰 변동이 없었으나 threshold detector의 평균반응단면적은 크게 변하였다. 따라서 속 중성자의 속 도분포에 관계없이 threshold detector의 평균반응단면적을 고정된 값으로 취하여 속 중성자선량을 계산한다면 큰 오차를 일으키게 될 것이라는 것을 보여주었다. 한편 핵 분열에서 방출되는 속 중성자의 속도분포에 대한 세 해석적 표현인 즉 Watt, Cranberg 및 Maxwellian 공식들로 부터 속 중성자선선량을 계산하여 서로 비교하였다. Watt 및 Cranberg 공식들로 부터 얻어진 속 중성자선량은 Maxwellian 공식으로 부터 얻어진 그것보다 약간 높은 값을 보여 주었으며 Watt 공식에 의한 선량계산치는 Cranberg 공식에 의한 그것과 비슷한 값을 보여 주었다.

## 1. Introduction

Owing to their well-known advantages-cheapness, adaptability, and simplicity of equipment—activation techniques[1-10] have been widely employed in measuring neutron energy spectra and neutron doses. The inaccuracy of differential cross-sections and responses of

*The author is now at Atomic Energy Research Institute, P.O. Box 7, Chungryang-ri, Seoul, Korea.

activation detectors over wide energy region, however, make it difficult to do a reliable interpretation for them from threshold-activation data.

Many attempts[2,11-18] have been made to interpret the data accurately by mathematical treatments, but no rigorous method has been found yet. Especially when one threshold detector is used in order to measure fast neutron doses, the interpretation is customarily based on the assumption[10] that the spectral distribution of fission neturons should approximate to that of the thermal induced $U^{235}$-fission. This convention could be adoptable in specially-designed facility such as the $U^{235}$-converter [11,19,20] from which a pure $U^{235}$-thermal fission spectrum may be obtained.

There is a question, however, whether it is correct to make a direct use of this convention under particular circumstances such as criticality accidents. In accidents of critical assemblies, there is a large possibility that fission can be due to all energy neutrons rather than thermal only. A number of investigators[21-27] reported that spectra varied with the incident neutron energy causing fission. If so, the spectral distribution of fission neutrons emitted from a supercritical accident will have a shape different from the $U^{235}$-thermal fission spectrum, and the commonly adopted convention may no longer be applicable to this case. When the interpretation of fast neutron doses in criticality accidents is performed with a few thershold detectors, this phenomenon has not so far been taken into account.

In this study, attention has thus been drawn to this fact with the purpose of improving the accuracy of fast neutron dose estimation by the use of two threshold detectors (indium and sulphur) in the Harwell criticality locket[10].

Assuming that the spectral distribution of fission neutrons would be governed by one spectral parameter, a calculation of the surface-absorbed dose has been done with the help of the digital computer Electrologica X8.

In addition, the doses computed from three well-known formulae for the fission neutron spectrum, i.e., Watt[28], Cranberg et al.[29] and Maxwellian forms, are compared.

## 2. Convention on the Fission Neutron Spectrum

Watt[28] and Cranberg et al.[29] have proposed the following semi-empirical formulae for the $U^{235}$+ n(thermal) fission neutron spectrum:

the Watt's formula:

$$N_w(E) = 0.484\ exp(-E)\ sinh(2E)^{1/2} \quad \cdots\cdots(1)$$

the analytical form by Cranberg et al.:

$$N_c(E) = 0.453\ exp(-E/0.965) \times sinh(2.29E)^{1/2} \quad \cdots\cdots(2)$$

the Maxwellian form:

$$N_m(E) = 0.77E^{1/2}\ exp(-0.775E) \quad \cdots\cdots(3)$$

It should be pointed out that: $N_w(E)$, $N_c(E)$ and $N_m(E)$ are the number of neutrons of energy $E$ per unit energy interval in Watt, Cranberg et al. and Maxwellian formulae, respectively, and $E$ is the neutron energy in MeV. The Maxwellian expression was perhaps first introduced by Cranberg et al., but it was reported that a good fit of experimental data was obtained in the region up to 9 MeV.

Several investigators[21-27] made it clear that the spectral distribution of fission neutrons depends on the incident neutron energy leading to fission. This may be a strong hint that the above analytical expressions have to be applied only in the case of the thermal-induced $U^{235}$ fission; for other spectra a different mathematical formula has to be used in accordance with the fission type.

In almost all cases of criticality accidents, the source neutrons causing fission can be fission neutron spectrum itself rather than thermal. The spectral distribution, therefore, may not be expressed by the same formulae as given in the above. In other words, this means that a

direct use of the above formulae regardless of the fission type may be subject to a considerable error in making the interpretation of fast neutron doses.

On this occasion, an attempt is therefore made finding the fission type in criticality accidents by introducing the spectral parameters from which the spectral variation may be deduced. It is achieved by making the above equations generalized as follows:

$$N_m(E)=K(\alpha,\beta)\ exp(-\alpha E)\ sinh\beta E^{1/2} \cdots\cdots (4)$$

$$N_w(E)=K(\alpha,\beta)E^{\alpha}\ exp(-\beta E) \cdots\cdots (5)$$

in which the generalized Cranberg form, $N_c(E)$, is the same as that of Watt, and $K(\alpha,\beta)$ is constant for normalizing to $\int_0^{\infty} N(E)\,dE=1$ where $N(E)$ represents a general expression for $N_w(E)$, $N_c(E)$ and $N_m(E)$. In this report, $\alpha$ and $\beta$ are employed as spectral parameters which might be able to determine the spectral shape of fission neutrons. These parameters except $\alpha$ in the Maxwellian form are constants basically related to the nuclear temperature of a fission fragment, and have in part an inverse relation to the nuclear temperature according to the work of Terrell[30]. He had theoretically analyzed experimental data reported by many authors[28,29] on the basis of an evaporation mechanism by which neutrons are emitted from fast-moving excited fission fragments, and then predicted variation of the nuclear temperature with the fission type. His theory is backed up by many experimenters[21-27]; the nuclear temperature increases with increasing incident energy of neutrons which cause fission. Since the nuclear temperature is linearly related to the average fission neutron energy, increase of the nuclear temperature will mean increase of the average fission neutron energy (hereinafter the average energy is to be understood as the average fission neutron energy). From the above, it may be said that the average energy of uncollided neutrons emitted from criticality accidents is higher than that in the thermal-induced $U^{235}$ fission. In the generalized forms [Eqs. (4) and (5)], the normalization constant $K(\alpha,\beta)$ and the average enregy $\bar{E}$ are:

for the Watt's and Cranberg's forms:

$$K(\alpha,\beta)=\frac{2\alpha^{3/2}}{\beta\pi^{1/2}}\ exp\left(-\frac{\beta^2}{4\alpha}\right) \cdots\cdots (6)$$

$$\bar{E}=\frac{3}{2\alpha}+\frac{\beta^2}{4\alpha^2}, \cdots\cdots (7)$$

for the Maxwellian form:

$$K(\alpha,\beta)=\frac{\beta^{\alpha+1}}{\alpha\Gamma(\alpha)} \cdots\cdots (8)$$

$$\bar{E}=\frac{\alpha+1}{\beta} \cdots\cdots (9)$$

The forms [Eqs. (4) and (5)] having two spectral parameters make it necessary to use at least three threshold detectors, but with compromise of our purpose for which two threshold detectors are available we intend to adopt the forms with one spectral parameter, so that one of two parameters should be given by a fixed value. For simplicity of mathematical approach the spectral parameter, $\alpha$, is fixed as given in Eqs. (1), (2) and (3), so the spectral distribution of fission neutrons may be governed by the spectral parameter $\beta$. Therefrom, Eqs. (4) and (5) will be formed:

for the Watt's formula:

$$N_w(E)=\frac{5,057}{(4\bar{E}-6)^{1/2}}\ exp\left[-(\bar{E}+E)\right] \times sinh\left[(4\bar{E}-6)E\right]^{1/2} \cdots\cdots (10)$$

for the Cranberg's formula:

$$N_c(E)=\frac{5,335}{(4,295\bar{E}-6,218)^{1/2}}\ exp[-1,036 \times(\bar{E}+E)]\ sinh[(4,295\bar{E}-6,218)E]^{1/2} \cdots (11)$$

for the Maxwellian formula:

$$N_m(E)=\frac{2,093}{\bar{E}}\left(\frac{E}{\bar{E}}\right)^{1/2} \times exp\left[-1.5\left(\frac{E}{\bar{E}}\right)\right] \cdots\cdots (12)$$

where all the symbols have the usual meaning as given elsewhere in this report. These forms imply that the average energy $\bar{E}$ could build up

a magnitude of the spectral deviation of fission neutrons.

## 3. Determination of Spectral Parameter

The spectral parameter $\beta$ (or the average energy $\bar{E}$ since they are closely related) is determined from measurments of the spectral indices [2,3,18,31] which are defined as the average cross-section ratio for threshold detectors A and B in the fission neutron spectrum. A detailed description about the spectral index is found in the excellent work of Grundl [18]. In the indium and sulphur threshold detector system, the $In^{115}(n,n')In^{115m}$ and $S^{32}(n,p)p^{32}$ nuclear reactions are applied for our purpose. In this detector system and our counting equipment which are calibrated for neutron dosimetry in criticality accidents [32], the spectral index is given by the form

$$S_{in\cdot s}\equiv\frac{{}_E\bar{\sigma}_{in}}{{}_E\bar{\sigma}_s}=0.279\frac{A_{in}}{A_s} \quad \cdots\cdots(13)$$

where $S_{in\cdot s}$ is the spectral index for the indium and sulphur threshold detector system, ${}_E\bar{\sigma}_{in}$ and ${}_E\bar{\sigma}_s$ are the average microscopic cross-sections ($cm^2$) with regard to the fission neutron spectrum with the average energy $\bar{E}$ for indium and sulphur respectively. $A_{in}$ and $A_s$ represent activities (in cpm obtained by our counters) in the indium and sulphur discs immediately after exposure, and the factor given in the right-hand side of Eq. (13) was calculated with consideration of isotopic abundance, radiation yield per disintegration, counting efficiency, decay constant, atomic mass and other physical constant such as weight of the threshold detectors used. As can be seen in Eq.(13), the spectral index is a measured and computable quantity linking the experimental results to conclusion from the mathematical models as expressed in Eqs.(10), (11) and (12) for the fission neutron energy spectrum. For our purpose, it is more convenient to know the spectral index-to-average cross-section relationship than the spectral index-to-average energy relationship. It is because by the latter the neutron spectral distribution only is known. Therefore, another tedious computation is necessary in order to know the reaction rate of threshold detector. However, from the former relationship the reaction rate of threshold detector can be directly determined. Furthermore, it includes the term of the average energy in itself because the average cross-section is obtained from the spectrum which is built up with an average energy $\bar{E}$. The interest of this work was therefore concentrated on observing variations of the average cross-sections and the spectral indices as the average energy $\bar{E}$ is changed. The average cross section ,$\bar{\sigma}$, is formed

$$\bar{\sigma}=\frac{\int_0^\infty \sigma(E)\ N(E)\,dE}{\int_0^\infty N(E)\,dE} \quad \cdots\cdots(14)$$

where $\sigma(E)$ is differential cross-sections and $N(E)$ is the mathematical models for the spectrum as given in Eqs. (10), (11) and (12).

With recently published data [33-38] on the threshold activation cross-sections, the calculations of the average cross-sections as well as the spectral indices have been performed by a computer code for the various values of the average energies $\bar{E}$ in the region 1.5 to 2.5 MeV and for the threshold detectors under investigation. The limits in the average energy were arbitrarily taken. Eqs.(10) and (11) become an imaginary function at the average energy below about 1.5 MeV, so $\bar{E}$ = 1.5 MeV was taken as the lowest limit in the calculations. In the case of Eq.(12) the lowest limit was chosen arbitrarily, but in order to keep comparative value with those in Eqs.(10) and (11). The upper limit was also chosen arbitrarily.

## 4. Interpretation of Dose

The surface-absorbed dose is adopted in this

study. This is generally termed as a dose on the basis of the multi-collision process of neutrons in the body, in contrast with that of the first collision or single collision within a small piece of tissue in free air at the position of exposure. At present, it can be rarely claimed that the former concept has more significant meaning than the latter in so far as the biological effect in criticality accidents is concerned. However, there is an advantage [10] that the dose obtained on this concept (surface-absorbed dose) is consistent with the routine interpretation of the gamma-ray film badge. At the same time, it has a merit that the dose from the external gamma-ray and from the process in the body is able to be readily measured at the body's surface by the conventional film dosimetry.

In the standpoint of radiation protective measures, it is very desirable to give the dose expressed in the term "rem" which implies that the absorbed dose (expressed in rads) should be multiplied by an appropriate weighting factor. It is hardly possible, however, to estimate the rem-dose because the quality factor with regard to acute radiation exposure such as in criticality accidents has so far been unknown. On this occasion, the concept of the absorbed-dose (rads) will thus be used preliminary and is expected to be reconsidered when the reliable data on the quality factor are available. Hereinafter the term "dose" is to be understood as "absorbed dose (rads)".

The surface-absorbed dose was calculated from the depth dose curve of Snyder and Neufeld [39]. They have theoretically computed the depth dose as a function of incident neutron energy from thermal to 10 MeV in an infinite slab of tissue-equivalent material which is 30cm thick, by applying a Monte Carlo-method. Although some works [40, 41] have so far been made to confirm experimentally the calculated values of Snyder and Neufeld, no reliable data got on the actual human-like phantom are available yet. Therefore, the dose interpretation is essentially based on the Snyder and Neufeld's depth dose curve up to 10 MeV. With respect to neutrons from 10 to 15 MeV no appropriate data are avialable, so in this region a fitting was achieved on the assumption that the depth dose curve would be a smoothly varying function with energy. By a least-squares method, the following relationships between the dose and the the neutron energy were obtained:

in the region thermal $<E<1.5\times10^{-3}$ *Mev:*

$$D_1(E)=1.988\times10^{-11}\ E^{-0.061} \quad \cdots\cdots(15)$$

in the region $1.5\times10^{-3}$ *Mev* $<E<1.1$ *Mev:*

$$D_2(E)=3.477\times10^{-9}\ E^{0.72} \quad \cdots\cdots(16)$$

in the region 1.1 *Mev* $<E<15$ *Mev:*

$$D_3(E)=3.152\times10^{-9}E^{0.313} \quad \cdots\cdots(17)$$

In these equations $D_1(E)$ $D_2(E)$, and $D_3(E)$ are the surface-absorbed dose [rads/(n/cm$^2$)] per unit energy interval in each energy region, and $E$ is the neutron energy in MeV,

If $D(E)$ is the function defined in the corresponding energy region by $D_1(E)$, $D_2(E)$ and $D_3(E)$, the dose which is expected from a unit fluence of the fission neutron spectrum, $D$ [rads/(n/cm$^2$)], is

$$D=\int_0^\infty D(E)\,N(E)\,dE \quad \cdots\cdots(18)$$

Practically a numerical integration was performed in the interval from the thermal up to 15 MeV. Although a contribution due to neutrons above 15 MeV can be expected, its portion compared to the region considered is small enough so as to be discarded.

From measurements of the activity induced in the threshold detectors by means of the calibrated counters, total surface-absorbed fast neutron dose, $D_T$* (in rads) with the ex-

*For a unit fluence of the fission neutron spectrum and for the activity, $A_c$, which is obtained by counters, $D_T$ is given $D_T=\dfrac{A_cD}{VN\bar{\sigma}\lambda}$

where $V$ is volume of threshold detector used, $N$ is number of atoms per cubic centimeter, $\bar{\sigma}$ is the average cross-section, $\lambda$ is decay constant, and in $A_c$ radiation yield per disintegration and counting efficiency should be included in order to know the absolute activity.

ception of the external gamma-ray and of the $H^1(n,\gamma)D^2$-reaction in the body, is then obtained[31]:

in accordance with the activity of sulphur discs:

$$D_T=\frac{A_S D}{8.11\times10^{16}\,{}_{\bar{E}}\bar{\sigma}_s} \quad \cdots\cdots(19)$$

in accordance with the activity of indium foils:

$$D_T=\frac{A_{in} D}{2.91\times10^{17}\,{}_{\bar{E}}\bar{\sigma}_{in}} \quad \cdots\cdots(20)$$

Here all the symbols are the same as given elsewhere in this report. The factors given in denominators of both equations (19) and (20) were calculated with consideration of isotopic abundnace, radiation yield per disintegration, counting efficiency, decay constant, atomic mass and weight of the threshold detectors used, and have $min^{-1}$ in their dimension.

## 5. Results and Discussions

In Figs. 1 and 2, the relation is graphically plotted as the spectral indices *vs* the average cross-sections computed for the indium and sulphur detectors, respectively. On the curves in Figs. 1 and 2 are indicated the upper and lowest limits chosen arbitrarily as well as the average energy relative to the thermal-induced $U^{235}$

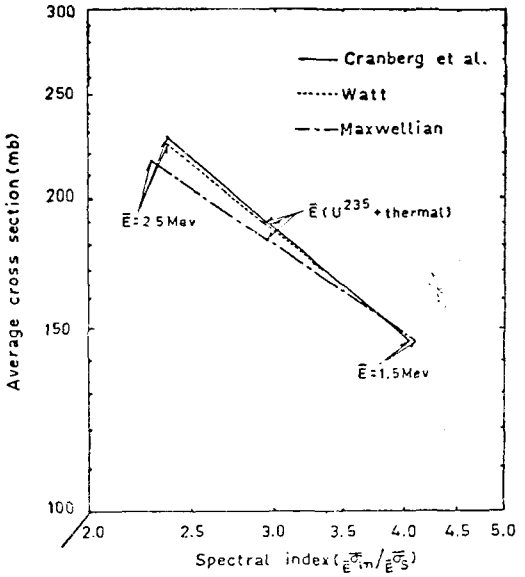


**Fig. 1. Variation of average cross section ($_{\bar{E}}\bar{\sigma}_{ni}$) with spectral index for the $In^{115}$ (n,n') $In^{115m}$ threshold reaction.**

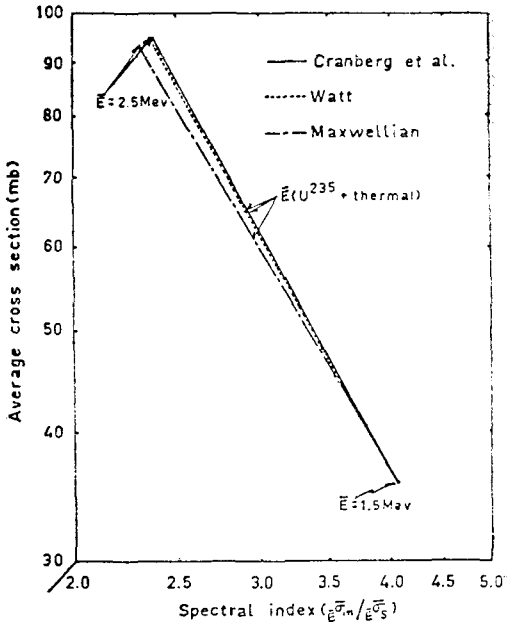


**Fig. 2. Variation of average cross section ($_{\bar{E}}\bar{\sigma}_{in}$) with spectral index for the $S^{32}$(n,p)$p^{32}$ threshold reaction.**

fission specturm.

As can be seen, the average cross-sections increase with decreasing spectral indices, a linearity being characterized on the log-log scale. In the same figures, the results based on the Maxwellian form show a departure from those on Watt and Cranberg et al. This tendency is also demonstrated in Table I for the $U^{235}$ + n (thermal) fission spectrum, and the reason will be considered later.

It has been discussed by many investigators[21-27] that the average energy $\bar{E}$ of fission neutrons increases with increasing incident neutron energy leading to fission. In other words, this means that no one could expect fission neutron spectra with having $\bar{E}$-values below the $\bar{E}$-value for the thermal-induced fission neutron spectrum, and that in our threshold detector system no spectra may be observed with values above the spectral index equivalent to $\bar{E}$ for the thermal fission spectrum as given in Table II.

Under particular circumstances such as super-critical accidents, fission may be induced by

**Table I. Average cross-sections* relative to the $U^{235}+n$ (thermal) fission neutron energy spectrum**

| $In^{115}(n,n')In^{115m}$ | | | $S^{32}(n,p)P^{32}$ | | |
|---|---|---|---|---|---|
| $\bar{\sigma}_{in}$ (mb) | Ref. | Remarks | $\bar{\sigma}_{s}$ (mb) | Ref. | Remarks |
| 168 | 42 | calculation | 65 | 9 | not specified |
| 174 | 13 | not specified | 63.8 | 13 | not specified |
| 170 | 9 | calculation | 65.7 | 4 | calulation using the Watt's form. |
| 181±10 | 43 | experiment | 60±1.2 | 19 | experiment |
| 200±10 | 44 | experiment. | 74±3 | 44 | experiment. |
| 182±20 | | calculation using the Watt's form. | 67±7 | | calculation using the Watt's form. |
| 201±21 | | calculation using the Grundl multi-group fit. | 79±8 | | calculaton using the Grundl multi-group fit. |
| 190.1 | present work (calculation) | using the Watt's form. | 64.9 | present work (calculation) | using the Watt's form. |
| 189.2 | | using the form of Cranberg et al. | 63.7 | | using the form of Cranberg et al. |
| 182.6 | | using the Maxwellian form. | 61.2 | | using the Maxwellian form. |

*For comparison data obtained by many workers are included together with that by the present work. Except those based on the early work of Martin et al.[33] in the case of indium, generally the results are in good agreement with others. There are still, of course, discrepancies between them. However, this deviation is not serious, if one takes into account that in calculation the data-fitting from the excitation curve published can not be protracted through always-consistence, and that in experiments there may be a source of error which can not be solved completely.

No attempt has been made to estimate the uncertainties in the average cross-sections because 1) values of the differential cross-sections are reported in many articles without an estimation of error, and 2) in the work in which errors are included, relative error bands show a spread in neutron energy as well as in the differential cross-sections, making an evaluation of the standard error difficult.

**Table II. Spectral index corresponding to the average energy E for the $U^{235}+n$ (thermal) fission neutron energy specrum**

| Analytical forms for the fission neutron spectrum | Average energy (MeV) | Spectral index |
|---|---|---|
| Watt | 2,000 | 2,928 |
| Cranberg et al. | 1,980 | 2,971 |
| Mexwellian | 1,935 | 2,984 |

neutrons with energies up to the maximum value, which exist in such critical assemblies. The spectrum will therefore make a shift towards values above the average energy for the thermal fission spectrum, and correspondingly it may give rise to an increase of the average cross-section along with a decrease of the spectral indices in the $In^{115}(n,n')In^{115m}$ and $S^{32}(n,p)P^{32}$ system.

Unfortunately it is very common that the fission neutron spectrum is contaminated by scattered neutrons, leading to very exceptional conditions, that is, the average cross-sections are lower than that for the thermal fission spectrum. In this case, the convention made here in this study is no more valid. In fact, it is hardly possible to determine the fission spectrum accurately by means of either one or a few threshold detector systems. More accurate information for the spectrum can of course be obtained from a use of multi-threshold detector system. In the practical point of view, however, it is customary to avoid multi-threshold detector sets for criticality dosimetry. In the case where a few threshold detectors are available, therefore, a compromise should be made. Even

if a degree of the accuracy is decreased, aquick-but to some extent reliable-dose estimation has first to be made after criticality accidents. From this, data fittings are carried out including values below the average cross-section as given in Table I. For convenience' sake, mathematical expressions are drawn providing the relation between the spectral index and the average cross-section with the aid of a least-squares fit. They are for the $In^{115}$ (n,n') $In^{115m}$ and $S^{32}$ (n,p) $p^{32}$ threshold reactions:

a) With regard to the Watt's form:

1) $$_{E}\bar{\sigma}_{in}=456\left(\frac{_{E}\bar{\sigma}_{in}}{_{E}\bar{\sigma}_{s}}\right)^{-0.815} \quad \cdots\cdots(21)$$

in conjunction with Eq. (13)

$$_{E}\bar{\sigma}_{in}=1289\left(\frac{A_{in}}{A_{s}}\right)^{-0.815} \quad \cdots\cdots(22)$$

2) $$_{E}\bar{\sigma}_{s}=456\left(\frac{_{E}\bar{\sigma}_{in}}{_{E}\bar{\sigma}_{s}}\right)^{-1.815} \quad \cdots\cdots(23)$$

in conjunction with Eq. (13)

$$_{E}\bar{\sigma}_{s}=4618\left(\frac{A_{in}}{A_{s}}\right)^{-1.815} \quad \cdots\cdots(24)$$

b) with regard to the formula of Cranberg et al.:

1) $$_{E}\bar{\sigma}_{in}=468\left(\frac{_{E}\bar{\sigma}_{in}}{_{E}\bar{\sigma}_{s}}\right)^{-0.832} \quad \cdots\cdots(25)$$

in conjunction with Eq. (13)

$$_{E}\bar{\sigma}_{in}=1353\left(\frac{A_{in}}{A_{s}}\right)^{-0.832} \quad \cdots\cdots(26)$$

2) $$_{E}\bar{\sigma}_{s}=463\left(\frac{_{E}\bar{\sigma}_{in}}{_{E}\bar{\sigma}_{s}}\right)^{-1.825} \quad \cdots\cdots(27)$$

in conjunction with Eq.(13)

$$_{E}\bar{\sigma}_{s}=4756\left(\frac{A_{in}}{A_{s}}\right)^{-1.825} \quad \cdots\cdots(28)$$

c) with regard to the Maxwellian espression:

1) $$_{E}\bar{\sigma}_{in}=385\left(\frac{_{E}\bar{\sigma}_{in}}{_{E}\bar{\sigma}_{s}}\right)^{-0.687} \quad \cdots\cdots(29)$$

in conjunction with Eq. (13)

$$_{E}\bar{\sigma}_{in}=926\left(\frac{A_{in}}{A_{s}}\right)^{-0.687} \quad \cdots\cdots(30)$$

2) $$_{E}\bar{\sigma}_{s}=382\left(\frac{_{E}\bar{\sigma}_{in}}{_{E}\bar{\sigma}_{s}}\right)^{-1.681} \quad \cdots\cdots(31)$$

in conjunction with Eq. (13)

$$_{E}\bar{\sigma}_{s}=3264\left(\frac{A_{in}}{A_{s}}\right)^{-1.681} \quad \cdots\cdots(32)$$

Here suffices *in* and *s* refer to the threshold detectors indium and sulphur, respectively, and other symbols have the same meaning as given elsewhere in this report. From the above simple formulae the average cross-sections may readily be determined.

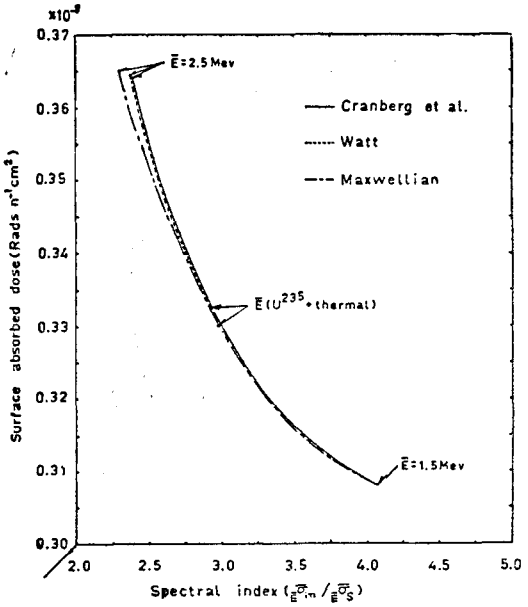


**Fig. 3. Surface absorbed dose as a function of spectral index.**

On the other hand, the results obtained from an integration of Eq. (18) with various values of the average energy $\bar{E}$ are shown in Fig. 3 which is a graph of the spectral index $_{E}\bar{\sigma}_{in}$ / $_{E}\bar{\sigma}_{s}$ against the surface-absorbed dose per unit fluence of the fission neutron spectrum. On the curves in Fig. 3 are indicated the upper and lowest limits taken arbitrarily as well as the average energy relative to the thermal-induced $U^{235}$ fission spectrum. As can be seen in Fig. 3, there is only a few percent deviation between Watt, Cranberg and Maxwellian forms. A mathematical expression governing the spectral index-to-the surface-absorbed dose conversion curve is thus derived from only the form of Cranberg et al., being the latest form reported. In view of practical application as discussed before, data fittings are carried out including values above the spectral index as depicted in Table II. The best curve fit is third degree

polynomial:

$$D=\left[9.299-4.5\left(\frac{\bar{E}\sigma_{in}}{\bar{E}\sigma_s}\right)+1,122\left(\frac{\bar{E}\sigma_{in}}{\bar{E}\sigma_s}\right)^2\right.$$
$$\left.-0.096\left(\frac{\bar{E}\sigma_{in}}{\bar{E}\sigma_s}\right)^3\right]10^{-9} rads/(n/cm^2) \quad \cdots\cdots(33)$$

in which $D$ is the surface-absorbed dose for a unit fluence of the fission neutron spectrum, and the spectral index $\bar{E}\sigma_{in}/\bar{E}\sigma_s$ is now the quantity which can be measured. In combination with Eq. (13) the above formula becomes

$$D=\left[9.299-1.256\left(\frac{A_{in}}{A_s}\right)+0.087\left(\frac{A_{in}}{A_s}\right)^2\right.$$
$$\left.-0.0021\left(\frac{A_{in}}{A_s}\right)^3\right]10^{-9} rads/(n/cm^2) \quad \cdots(34)$$

Therefrom, the total surface-absorbed fast neutron dose for a count rate of A cpm from the threshold detectors immediately after exposure will be calculated, and convenient formulae are given in association with Eqs. [(19) — (20)] and Eqs. [(21) — (34)]:

a') for the Watt's form:

1) from the indium foils

$$D_T=A_{in}\left(\frac{A_{in}}{A_s}\right)^{0.815}\left[2.48-3.35\times10^{-1}\left(\frac{A_{in}}{A_s}\right)\right.$$
$$\left.+2.32\times10^{-2}\left(\frac{A_{in}}{A_s}\right)^2-5.6\times10^{-4}\left(\frac{A_{in}}{A_s}\right)^3\right]10^{-2}$$
$$rads \quad \cdots\cdots(35)$$

2) from the sulphur discs

$$D_T=A_s\left(\frac{A_{in}}{A_s}\right)^{1.815}\left[2.483-3.354\times10^{-1}\left(\frac{A_{in}}{A_s}\right)\right.$$
$$\left.+2.323\times10^{-2}\left(\frac{A_{in}}{A_s}\right)^2-5.61\times10^{-4}\left(\frac{A_{in}}{A_s}\right)^3\right]10^{-2}$$
$$rads \quad \cdots\cdots(36)$$

b') for the form of Cranberg et al.:

1) from the indium foils

$$D_T=A_{in}\left(\frac{A_{in}}{A_s}\right)^{0.832}\left[2.36-3.19\times10^{-1}\left(\frac{A_{in}}{A_s}\right)\right.$$
$$\left.+2.21\times10^{-2}\left(\frac{A_{in}}{A_s}\right)^2-5.33\times10^{-4}\left(\frac{A_{in}}{A_s}\right)^3\right]10^{-2}$$
$$rads \quad \cdots\cdots(37)$$

2) from the sulphur discs

$$D_T=A_s\left(\frac{A_{in}}{A_s}\right)^{1.825}\left[2.41-3.26\times10^{-1}\left(\frac{A_{in}}{A_s}\right)\right.$$
$$\left.+2.26\times10^{-2}\left(\frac{A_{in}}{A_s}\right)^2-5.44\times10^{-4}\left(\frac{A_{in}}{A_s}\right)^3\right]10^{-2}$$
$$rads \quad \cdots\cdots(38)$$

c') for the Maxwellian form:

1) from the indium foils

$$D_T=A_{in}\left(\frac{A_{in}}{A_s}\right)^{0.687}\left[3.45-4.66\times10^{-1}\left(\frac{A_{in}}{A_s}\right)\right.$$
$$\left.+3.23\times10^{-2}\left(\frac{A_{in}}{A_s}\right)^2-7.80\times10^{-4}\left(\frac{A_{in}}{A_s}\right)^3\right]10^{-2}$$
$$rads \quad \cdots\cdots(39)$$

2) from the sulphur discs

$$D_T=A_s\left(\frac{A_{in}}{A_s}\right)^{1.681}\left[3.51-4.75\times10^{-1}\left(\frac{A_{in}}{A_s}\right)\right.$$
$$\left.+3.29\times10^{-2}\left(\frac{A_{in}}{A_s}\right)^2-7.93\times10^{-4}\left(\frac{A_{in}}{A_s}\right)^3\right]10^{-2}$$
$$rads \quad \cdots\cdots(40)$$

With these equations, the fast neutron dose will be determined and for the same expression the dose obtained from the sulphur discs will be equal to that from the indium foils. All the above formulae [Eqs. (35) — (40)] are given for comparison. The Maxwellian form will yield a smaller dose than the Watt's and Cranberg's forms, both being similarly close. The reason for this deviation would be due in part to the fact that although all the expressions [Eqs. (1), (2) and (3)] were basically established on the same assumptions*, semi-empirical forms proposed by Watt[28] and Cranberg et al.[29] were derived from an analytical fitting in favor of experimental data.

By simply assuming that the spectrum is governed by one spectral parameter taken on the mathematical basis rather than the physical one, the above convenient equations [Eqs. (35) — (40)] were obtained. At present, there is no way of claiming that this method presented here can give more reliable dose compared to the customary convention as set out previously. Although a discussion should follow by testing experimentally its reliability in various fission neutron fields, it was revealed that this convention is satisfactory by a preliminary test. The

*These assumptions are:
1) neutrons are emitted from moving fission fragments,
2) isotropic emission of neutrons in the center of mass system.

test was performed in a $U^{235}$-converter facility [45] which is designed for a purpose of biological research, and where data on the spectrum are available. The results showed quite a good agreement with the experimental data obtained by Davids et al. [45]

One might say that an application of the two-parameters form will give more reliable results than the one-parameter form. There is no evidence, however, that a best fit relative to fission spectra can be obtained by the forms with two parameters, except that from the mathematical point of view it gives a better description of fission neutron spectra. It should be noted that the formulae proposed by Watt and Cranberg et al. were constructed on the simple assumption, namely, neutrons from a fission fragment are isotropically given off in the center of mass system. As suggested in many articles [21, 30, 46], neutrons are not isotropically emitted from fission fragments even in the case of thermal fission. In other words, this may be a suggestion that in the laboratory system of neutron emission the spectrum is expressed with a more complicated mathematical form.

Even if it is assumed that the spectrum is governed by two parameters, it is liable to lose its significance under particular circumstances such as accidental excursions of critical assemblies, where a contamination due to scattered neutrons is unable to be eliminated. In addition, body scattered neutrons would disturb the incident neutron spectrum when activation detectors were worn on the body as a personnel dosimeter. It may be stated, therefore, that a reliable interpretation of the fast neutron dose in criticality accidents is hardly possible by using either one or a few threshold detector systems because the spectrum shape has to be assumed.

## 6. Conclusion

In neutron environments where the initial fission neutron spectrum is free from contamination due to scattered neutrons, the fast neutron dose with a few threshold detectors may be reasonably obtained by establishing the interpretation method based on the convention that the fission is able to be resulted from neutrons with energies up to the maximum value, where exist in the critical assembly.

In many cases of criticality accidents, however, where a lot of factors exist leading to a distortion of the initial fission neutron spectrum, it may unavoidably be necessary to employ a multi-threshold detector system if the accurate dose estimation is essentially emphasized.

## Acknowledgements

This work was performed during my stay in Rreactor Centrum Nederland, Petten, The Netherlands.

The author is deeply grateful to Dr. J.G. Ackers for his never-ending criticism and guidance throughout this work. My special word of thanks should be expressed to Dr. A. Glass who has rendered untiring assistance in a derivation of some mathematical formulae as well as in a preparation of the computer programme, and also for his fruitful discussions. My thanks are also due to Mr. J.G. Kamping who has kindly performed the computer calculations.

## References

1) J. B. Trice, Nucleonics, *16*, 81 (1958).
2) J. A. Grundl and A. Usner, Nucl. Sci. Eng., *8*, 598 (1960).
3) J. A. Grundl, Nucl. Sci. Eng., *31*, 191 (1968).
4) M. Bresesti, A.M. Del Turco, A. Ostidich, A. Rota and G. Segre, Neutron Dosimetry, *1*, IAEA, Vienna, 27 (1963).
5) G. I. Coulburn and T. G. Williamson, Nucl. Sci. Eng., *35*, 376 (1969).
6) G. S. Hurst, J. A. Harter, P. N. Hensley, W. A. Mills, M. Slater and P. W. Reinhardt, Rev.

Sci. Inst. *27*, 153 (1956).
7) J. Braun and R. Nilsson, Selected Topics in Radiation Dosimetry, IAEA, Vienna, 623 (1961).
8) F. J. Davis, Selected Topics in Radiation Dosimetry, IAEA, Vienna, 399 (1961).
9) W. G. Cross, Neutron Dosimetry, *1*, IAEA, Vienna, 389 (1963).
10) J. A. Dennis, AERE-R-report 4365 (1964).
11) D. J. Hughes, Pile Neutron Research, Cambridge (Mass.), Addison-Wesley Pub. Co., 1953.
12) J. B. Trice, APEX-report 408 (1957).
13) R. Dierckx, Neutron Dosimetry, *1*, IAEA, Vienna 325 (1963).
14) R. Gold, Nucl. Sci. Eng., *20*, 493 (1964).
15) G. Di Cola and A. Rota, Nucl. Sci. Eng., *23* 344 (1965).
16) W. N. McElroy, S. Berg and G. Gigas, Nucl. Sci. Eng., *27*, 533 (1967).
17) J. C. Ringle, UCRL-report 10732 (1963).
18) J. A. Grundl, LAMS-report 2883 (1963).
19) J. W. Boldeman, J. Nucl. Energy, Part A/B, *18*, 417 (1964).
20) A. M. Bresesti, M. Bresesti and R. A. Rydin, Nucl. Sci. Eng., *29*, 7 (1967).
21) R. B. Leachman, Phys. Rev., *101*, 1005 (1956).
22) R. B. Leachman, Proc. Int. Conf. Peaceful Uses of Atomic Energy, *2*, U.N., New York, 193 (1956).
23) I. I. Bondarenko, B. D. Kuzminov, L. S. Kutsayeva, L. I. Prokhorova and G. N. Smirenkin, Proc. 2nd. U.N. Conf. Peaceful Uses of Atomic Energy, *15*, Geneva, 353 (1958).
24) J. C. Hopkins and B. S. Diven, Nucl. Phys., *48*, 433 (1963).
25) D. S. Mather, P. Fieldhouse and A. Moat, Phys. Rev., *133*, 6B, B 1403 (1964).
26) E. Barnard, A. T. G. Ferguson, W. R. McMurray and I. J. Van Heerden, Nucl. Phys., *71*, 228 (1965).
27) H. Conde and G. During, Arkiv für Fysik, *29*, 313 (1965).
28) B. E. Watt, Phys. Rev., *87*, 1037 (1952).
29) L. Cranberg, G. Frye, N. Nereson and L. Rosen, Phys. Rev., *103*, 662 (1956).
30) J. Terrell, Phys. Rev., *113*, 527 (1959).
31) S. G. Ro, RCN-Int. report 70-004, Reactor Centrum Nederland, The Netherlands (1970)
32) S. G. Ro, RCN-Int. report 69-089, Reactor Centrum Nederland, The Netherlands (1969).
33) H. C. Martin, B. C. Diven and R. F. Taschek, Phys. Rev., *93*, 199 (1954).
34) H. O. Menlove, K. L. Coop and H. A. Grench, Phys. Rev., *163*, 1308 (1967).
35) E. D. Klema and A. O. Hanson, Phys. Rev., *73*, 106 (1948).
36) P. F. Rago and N. Goldstein, Health Phys., *14*, 595 (1968).
37) L. Allen Jr., W. A. Biggers, R. J. Prestwood and R. K. Smith, Phys. Rev., *107*, 1363 (1967).
38) H. Liskien and A. Paulsen, Compliation of cross-sections for some neutron induced threshold reactions, EUR 119.e, European Atomic Energy Community-Euratom, Geel, Belgium, 1963.
39) National Bureau of Standards, Protection against Neutron Radiation up to 30 Million Electron Volt, Handbook 63, 1957.
40) J. W. Smith and S. J. Boot, AERE-R report 3696 (1961).
41) C. E. Clifford, Health Phys., *15*, 527 (1969).
42) H. A. Hansen and J. M. Doderlein, Proc. 2nd U.N. Int. Conf. Peaceful Uses of Atomic Energy, *14*, 455 (1958).
43) W. Köhler and K. Knopf, Nukleonik, *10*, 181 (1967).
44) A. Fabby, Nukleonik, *10*, 280 (1967).
45) J. A. G. Davids, A. P. J. Mos and A. de Oude, Phys. Med. Biol., *14*, 573 (1969).
46) J. Terrell, Phys. Rev., *127*, 880 (1962).