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Fast Neutron Dosimetry in Nuclear Criticality Accidents

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ABSTRACT

The neutron dosimetrical parameters, i. e., the fission neutron spectrum-averaged cross-sections and the fluence-to-dose conversion factors have been calculated for some threshold detectors with the aid of a computer. The threshold detectors under investigation were the $^{115}In(n,n')^{115m}In$, $^{32}S(n,p)^{32}P$ and $^{27}Al(n,\alpha)^{24}Na$ reactions.

It is revealed that the average cross-sections($\bar{\sigma}$) for the $^{32}S(n,p)^{32}P$ reaction are independent of the spectral functions, namely, the Watt-Cranberg and Maxwellian forms. In the case of the $^{27}Al(n,\alpha)^{24}Na$ reaction a variation of the $\bar{\sigma}$ values appears to be highly dependent on the fissioning types.

It seems that both the average cross-section for the $^{115}In(n,n')^{115m}In$ reaction and the conversion factor are insensitive to the spectral deformation of fission neutrons. These phenomena make it applicable to use indium as a possible integral fast neutron dosimeter in nuclear criticality accidents provided that the virgin fission neutrons are completely free from the scattered neutrons.

1. Introduction

It is of importance to have a satisfactory system of fast neutron dosimetry, which can provide rapid dose estimates for exposed persons in the event of nuclear criticality accidents. The fast neutron dosimetry in such circumstances is commonly performed by the system based on threshold activation because of many desirable properties, i. e., simplicity, adaptability and the negligible gamma-sensitivity.

Hurst et al. [1] have developed a sophisticated system consisting of multi-component activation detectors. This system is relatively expensive for the practical use.

Ing and Cross[2] have proposed a single

component detector using the $^{103}Rh(n,n')^{103m}Rh$ reaction as a fast neutron personal dosimeter. It is reported that the dosimeter can yields dose reading effectively independent of virgin fission neutron spectrum. However, it is not desirable to adopt this dosimeter of which the radioactivity induced in ^{103}Rh is difficult to be absolutely measured and that is not readily available.

The purpose of this work is to examine the dosimetrical characteristics (or parameters) of some threshold detectors which may be components for a criticality accident dosimetry system, and is then to suggest indium as a possible integral dosimeter for measuring the fast neutron dose in nuclear criticality accidents under the scattered-free condition. The interesting dosimetrical parameters are the fission neutron spectrum-averaged cross-sections (hereinafter this terminology will be denoted by the average cross-sections) for the $^{115}In(n,n')^{115m}In$, $^{32}S(n,p)^{32}P$ and $^{27}Al(n,\alpha)^{24}Na$ reactions as well as the neutron fluence-to-dose conversion factors. These parameters have been numerically calculated by means of an electronic computer.

2. Calculation of Dosimetrical Parameters

The average cross-sections, $\bar{\sigma}$, is usually defined by

$$\bar{\sigma}=\int_0^{\infty}\sigma(E)N(E)dE\Big/\int_0^{\infty}N(E)dE \quad \cdots\cdots\cdots\cdots\cdots\cdots (1)$$

where $\sigma(E)$ is the diffrential cross-sections of the threshold activation reactions and N(E) the normalized spectral function of fission neutrons. The cross-section data for the $^{115}In(n,n')^{115m}In$, $^{32}S(n,p)^{32}P$ and $^{27}Al(n,\alpha)^{24}Na$ reaction were taken from a lot of articles. [3)-7)]

The normalized fission neutron or fast neutron spectrum, N(E), is expressed by two representative formulae, [8), 9)]

$$N_c(E)=\frac{0.655}{(0.667\bar{E}_c-0.493)^{1/2}}\exp\left[\frac{-(0.74+E)}{0.667\bar{E}_c-0.493}\right]\times\sinh\left(\frac{1.722E^{1/2}}{0.667\bar{E}_c-0.493}\right) \quad \cdots\cdots (2)$$

and

$$N_m(E)=\frac{2.073}{\bar{E}_m}\left(\frac{E}{\bar{E}_m}\right)^{1/2}\exp\left(-1.5E/\bar{E}_m\right) \quad \cdots\cdots (3)$$

in which subscipts c and m stand for the Watt-Cranberg[10), 11)] and Maxwellian[11)] forms, respectively. E is the neutron energy in MeV and $\bar{E}$ the average fission neutron energy (through-out this work this terminology will be represented by the average energy) that may be used as an adjustable parameter depending on the fissioning types in criticality accidents. A detailed description is made in the papers of Terrell[8)] and Ro et al.[9)]

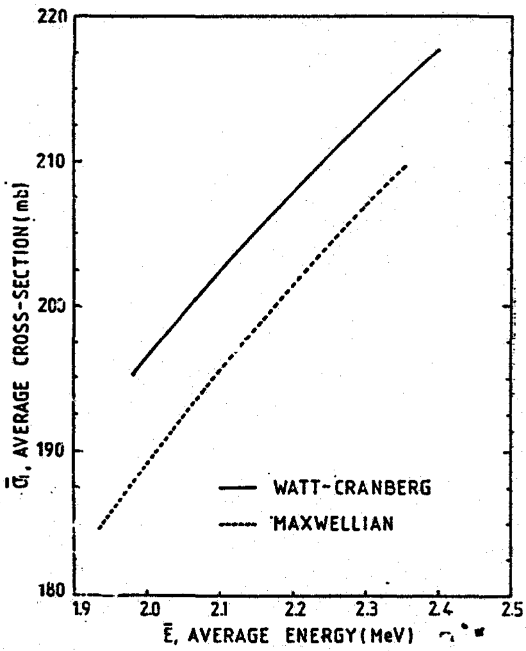

Fig. 1. Average cross-section of $^{115}In(n,n')^{115m}In$ reaction as a function of average fission neutron energies.

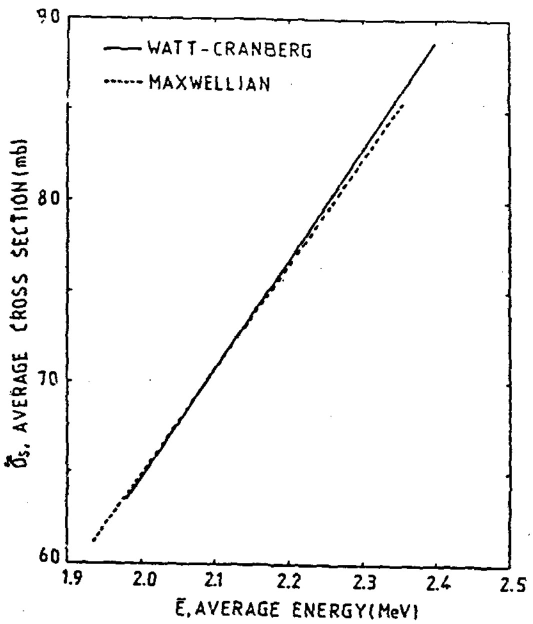

Fig. 2. Average cross-sections of $^{32}S(n,p)^{32}P$ reaction as a function of average fission neutron energies.

The average cross-sections of the threshold detectors under consideration were numerically calculated for the various values of the average energy $(\bar{E})$ in Eqs. (1) and (2) using an electronic computer CYBER-73. Practically a numerical calculation was performed in the interval from the thermal energy up to 20 MeV of neutron energy. Although the spectral function extends theoretically to infinite energy, only a vanishingly small contribution due to neutrons above 20 MeV is expected so as to be discarded. The average energy $\bar{E}$ values were varied from that corresponding to thermal-induced fission [1.98 Mev in Eq. (1) and 1.935 MeV in Eq. (3)] up to that of 20 MeV neutron induced-fission [2.40 MeV in Eq. (1) and 2.335 MeV in Eq. (2)][12), 13)].

The fluence-to-dose conversion factor was obtained by integrating the product of the differential dose and the fission neutron spectrum, that is, Eqs. (2) and (3). We expesed that

$$\bar{d}=\int_0^{\infty}(E)N(E)dE\Big/\int_0^{\infty}N(N)dE \qquad (4)$$

Here d(E) is the differential dose in terms of kerma and maximum dose, and was mainly extracted from reports.[14), 15)]

A numerical computation of $\bar{d}$ values was carried out by the computer. It should be noted that the autogamma dose from the $^1H(n,\gamma)^2D$ reaction in the human body was exmpted from the calculation of $\bar{d}$ based on the maximum dose, since it can be easily obtained by the conventional film dosimetry.

3. Results and Discussion

In Figs. 1-3 shown are the calculated average cross-sections as a function of the average energy $\bar{E}$ for the threshold reactions of interest. The solid curves are for the

Table 1. Comparison of average cross-sections ratio obtained by different spectral functions of fission neutrons

Fission type	Average Cross-Section Ratio		
	$\bar{\sigma}_{JW}/\bar{\sigma}_{JM}$	$\bar{\sigma}_{SW}/\bar{\sigma}_{SM}$	$\bar{\sigma}_{AW}/\bar{\sigma}_{AM}$
Thermal fission	1.057	1.037	0.694
20 MeV Neutron Eission	1.038	1.041	0.881
$\bar{\sigma}_W/\bar{\sigma}_M$ (for 20 MeV) / $\bar{\sigma}_W/\bar{\sigma}_M$ (for thermal)	0.982	1.004	1.270

Table 2. Average cross-sections of the threshold reactions for neutron spectra from ^{235}U+n (thermal) and+n (20MeV) Fissions

Fission type	Average Cross-Sections (mb)					
	Watt-Cranberg form			Maxwellian form		
	^{115}In (n, n')	^{32}S (n, P)	^{27}Al (n, α)	^{115}In (n, n')	^{32}S (n, P)	^{27}Al (n, α)
Thermal fission	195. 14	63. 47	0. 50	184. 62	61. 19	0. 72
20 MeV Neutron Fission	217. 63	88. 75	1. 63	209. 69	85. 27	1. 85
$\bar{\sigma}$ (for 20 MeV) / $\bar{\sigma}$ (for thermal)	1. 12	1. 40	3. 21	1. 14	1. 39	2. 57

Watt-Cranberg while the dashed curves are for the Maxwellian form. The subscripts I, S, and A denoted in the average cross-sections refer to indium, sulphur and aluminum, respectively. As can be seen in these figures, the average cross-sections increase with increasing the $\bar{E}$ values. The Watt-Cranberg form gives the $\bar{\sigma}$ values higher than the Maxwellian form for indium while for aluminum the former gives the $\bar{\sigma}$ values lower than the latter. This may be due in part to the fact that the Maxwellian spectrum is harder than the Watt-Cranberg spectrum. The interesting result is that the $\bar{\sigma}$ values for the ^{32}S (n, p) ^{32}P reaction are nearly independent of the spectal function, i. e., Watt-Cranberg and Maxwellian forms. This tendency is also demonstrated in Table 1. The suffices W and C in $\bar{\sigma}$ in Table 1 mean the Watt-Cranberg and Maxwellian forms, respectively.

In Table 2 summarized are the average cross-sections weighted for the neutron spectum from ^{235}U-fission induced by thermal and 20 MeV neutrons. In the last low of Table 2 included are the average cross-section ratios of the thermal-to-20 MeV neutron fissions.

For the ^{27}Al (n, α) reaction, the cross-section ratio is large in comparison with both ^{115}In (n, n') reactions.

This may hint that the $\bar{\sigma}$ values for the ^{27}Al (n, α) reaction are highly sensitive to the spectral deformation, subsequently allowing it to be used for obtaining a more elaborate spectral information of fission neutrons in nuclear citicality accidents.

In the case of the ^{115}In (n, n') reaction, the cross-section ratio is comparatively small, namely, 1. 12 and 1. 14 for the respective Watt-Crancerg and Maxwellian forms. In fact this ratio may be further deceased in actual circumstances of nuclear criticality accidents. Bondarenko et al.[16] report that the $\bar{E}$ value of fission neutrons induced by fission neutron itself was obsered to be 2. 03 MeV, leading to 1. 067 of the cross-section ratio for both Watt-Cranberg and Maxwellian forms. This may indicate that indium can yield neutron fluence values being effectively independent of virgin neutron spectrum in criticality accidents where the fission neutrons are completely free from contamination of the scattered neutrons.

In Table 3 presented the average cross-sections for the thermal-induced ^{235}U fission neutron spectrum, and the data obtained by many authors[17)-22)] are included for comparison. With the exception of those based on the early works in the case of indium, generally the result got in this study is in good agre-

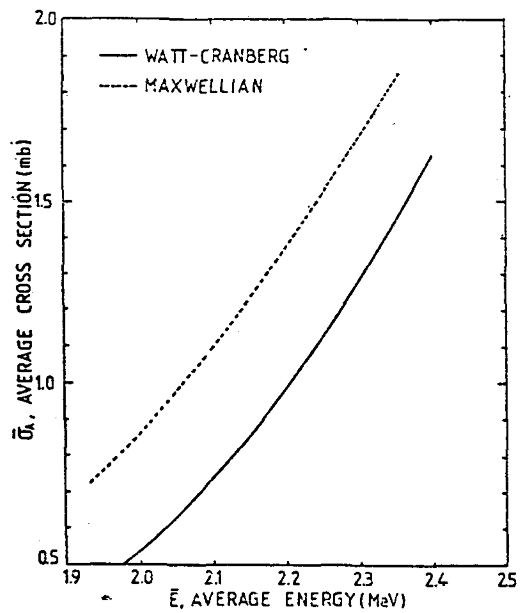

Fig. 3 Average cross-section of $^{27}Al(n,\alpha)^{24}Na$ reaction as a function of average fission neutron energies.

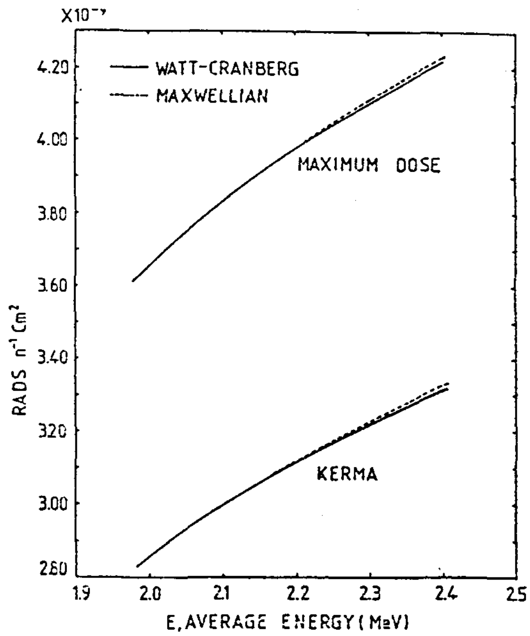

Fig. 4. Neutron fluence-to-dose conversion factor as a function of average fission neutron eneries.

Table 3. Average cross-sections relative to ^{235}u+n(thermal) fission neutron spectrum

$^{115}In(n,n')^{115m}In$			$^{32}S(n,p)^{32}P$			$^{27}Al(n,\alpha)^{24}Na$		
$\bar{\sigma}_I$(mb)	Ref.	Remarks	$\bar{\sigma}_S$(mb)	Ref.	Remarks	$\bar{\sigma}_A$(mb)	Ref.	Remarks
170	18	Calculation	95.7	17	Calculation	0.6	18	Calculation
174	19	Not specified	65	18	Not specified	0.59	19	Not specified
181±10	20	experiment	63.8	19	Not specified	0.60±0.03	20	experiment
200±10	22	experiment	60±1.2	20	experiment	0.78±0.03	22	experiment
195.14	this work	Calculation (for Watt-Cranberg)	63.47	this work	Calculation (for Watt-Cranberg)	0.5	this work	Calculation (for Watt Cranberg)
184.62	this work	Caluation (for Maxwellian)	61.19	this work	Calculation (for Maxwellian)	0.72	this work	Calculation (for Maxwellian)

ement with others. For the $^{27}Al(n,\alpha)^{24}Na$ reaction there is a large scatter between investigator. This scatter is a clear indication of some errors still to be isolated. The value got for the Watt-Cranberg form in this work is comparatively lower than that of the Maxwellian form and those by others.[18)-20),22)] As set out previously, this may in part come from the fact that the neutron spectrum of the former is soft compared to the latter.

Shown in Fig. 4 are the fluence-to-dose conversion factors as a function of the average energies. The solid lines are those for

Table 4. Computed kerma and maximum dose conversion factor per unit fluence for neutron spectra from ^{235}U+n (thermal) and+n (20MeV) fission

Fission type	Conversion factor			
	Watt-Cranberg		Maxwellian	
	Kerma (rads/(n/cm²))	Max. dose (rads/(n/cm²))	Kerma (rads/(n/cm²))	Max. dose (rads/(n/cm²))
n (thermal) fission	2.83×10^{-9}	3.61×10^{-9}	2.75×10^{-9}	3.52×10^{-9}
n (20MeV) fission	3.31×10^{-9}	4.22×10^{-9}	3.27×10^{-9}	4.16×10^{-9}
$\frac{\bar{d}\text{ (for 20 MeV)}}{\bar{d}\text{ (for thermal)}}$	1.17	1.17	1.19	1.19

the dose based on the Watt-Craberg form while the dashed lines are for the Maxwellian form. The dose conversion factor is insensitive to spectral functions together with the fissioning types. This is also listed in Table 4.

Taking account of the result that the $\bar{\sigma}$ values for the ^{115}In(n, n′) reaction is relatively independent of the fissioning types under consideration, this fact makes indium applicable as an integral fast neutron dosimeter in nuclear criticality accidents.

4. Conclusion

Neutron dosimetric parameters, i.e., the average cross-sections and the dose conversion factors were numerically calculated for ^{115}In(n, n′)^{115m}In, ^{32}S(n, p)^{32}P and ^{27}Al(n, α)^{24}Na reactions with the aid of a computer. As a conclusion, the followings can be drawn up:

1) The $\bar{\sigma}$ values for the ^{32}S(n, p)^{32}P reaction are independent of spectral functions, that is, the Watt-Cranberg and Maxwellian forms,

2) A variation of the $\bar{\sigma}$ values for the ^{27}Al(n, α) reaction is highly depending on the fissioning types. Therefrom, this reaction can be effectively used for determining the spectral deformation of fission neutrons,

3) Both $\bar{\sigma}$ for the ^{115}In(n, n′) reaction and $\bar{d}$ are insensitive to the fissioning types. It is, therefore, conclud ed that indium can be successfully used as a fast neutron dosimeter in criticality accidents where the virgin fission neutrons are completely free from the scattered neutrons.

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——抄 錄——

核臨界事故時에 있어서 速中性子線量 測定

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槪 要

여러가지 核分裂中性子 스펙트럼에 $^{32}S(n,p)$, $^{27}Al(n,\alpha)$ 및 $^{115}In(n,n')$의 勵起函數를 增率시켜 平均核反應斷面積을 電子計算機로 計算하였다.

그 結果 發端에너지가 높을수록 中性子스펙트럼 變化에 따라 平均 斷面積은 敏感하게 變化한다는 것이 判明되었다. 發端에너지가 比較的 낮은 인디움의 境遇, 核分裂特性에 따라 그의 平均 斷面積은 크게 變化되지 않았는데 中性子 散亂作用에 依한 影響이 排除될 수만 있다면 인디움은 核臨界事故時에 放出되는 中性子의 積算計로서 效果的으로 使用될 수 있을 것 같았다. 더욱이 中性子線量換算因子가 核分裂 中性子스펙트럼에 거의 無關하다는 事實은 인디움을 核臨界事故時의 中性子線量積算計로 사용할 수 있음을 뒷받침하는 것 같았다.

(但 上記 論文은 1977年 4月 24日 부터 4月 29日 까지 佛蘭西 파리에서 開催된 第4次 國際放射線防禦學會 및 總會(IRPA)에서 韓國代表로 參席하여 發表한 論文의 一部임.)