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**ATTENUATION DATA OF POINT ISOTROPIC NEUTRON  
SOURCES UP TO 400MeV IN WATER,  
ORDINARY CONCRETE AND IRON**

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**Hiroshi KOTEGAWA, Shun-ichi TANAKA, Yukio SAKAMOTO  
Yoshihiro NAKANE and Hiroshi NAKASHIMA**

**日本原子力研究所  
Japan Atomic Energy Research Institute**

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Attenuation Data of point Isotropic Neutron Sources  
up to 400MeV in Water, Ordinary Concrete and Iron

Hiroshi KOTEGAWA, Shun-ichi TANAKA, Yukio SAKAMOTO  
Yoshihiro NAKANE and Hiroshi NAKASHIMA

Department of Reactor Engineering  
Tokai Research Establishment  
Japan Atomic Energy Research Institute  
Tokai-mura, Naka-gun, Ibaraki-ken

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A comprehensive attenuation data of dose equivalent for point isotropic monoenergetic neutron sources up to 400MeV in infinite shields of water, ordinary concrete and iron has been calculated using the ANISN-JR code and a neutron-photon multigroup macroscopic cross section HIL086R. The attenuation factors were fitted to a 4th order polynomial exponent formula, making possible to use easily for point kernel codes. Additional data in finite shielding geometry was also calculated to correct the effect due to infinite medium, giving the maximum correction of 0.23 in the region for more 400 cm distance from neutron source of 400 MeV in iron shield.

Effective attenuation length for monoenergetic neutrons have been studied in detail. Subsequently, it was shown that the attenuation length was strongly dependent upon the penetration length and the Moyer's formula using a single attenuation length brought large error into the dose estimation behind thick shields for the intermediate energy neutrons up to 400 MeV. Furthermore, it was demonstrated that there was difference more than 50 % in the attenuation length of iron between the calculations with HIL086R and HIL086 because of the self-shielding effect.

Keywords : Attenuation Data, High Energy Neutron Source, Secondary Gamma-Rays,  
Point Isotropic Source, Dose Equivalent, Water, Concrete, Iron,  
HIL086R, HIL086, Attenuation Length

水、コンクリート、鉄中での400MeVまでの  
点等方中性子線源からの減衰データ

日本原子力研究所東海研究所原子炉工学部

小手川 洋・田中 俊一・坂本 幸夫・中根 佳弘・中島 宏

(1994年5月31日受理)

ANISNコードと中性子- $\gamma$ 線多群巨視的断面積H I L O 8 6 Rを使って、400MeVまでの点等方単色エネルギー中性子を線源とする無限厚さの水、普通コンクリート、鉄の各単層遮蔽体に対する中性子と2次 $\gamma$ 線の線量当量減衰係数を計算した。点減衰核簡易計算コードPKN-Hで利用するため、4次の多項式を含む指数関数で近似した。無限媒質による反射の効果を補正するための補足的データも評価した。反射の効果は400MeV中性子線源に対し鉄遮蔽体中400cm以上で最大0.23に達する。

単色中性子に対する実効的減弱距離を詳細に検討した。減弱距離は透過距離に強く依存し、単一の減弱距離を用いるモイヤーモデルの式は、少なくとも400MeVまでの中間エネルギー中性子に対する遮蔽体透過後の線量評価に大きな誤差が生じることを示した。

さらに、自己遮蔽効果を考慮しているH I L O 8 6 Rは自己遮蔽効果を考慮していないH I L O 8 6 と比べて、鉄の減弱距離は、50%以上大きくなる。

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## 1. INTRODUCTION

The present authors have calculated the attenuation data<sup>1)</sup> of neutron and secondary gamma-ray dose equivalents for neutron sources from 0.01 MeV up to 14.9 MeV in water, ordinary concrete and iron shields based on the JSD-100 library,<sup>2)</sup> and developed a point kernel code PKN<sup>3)</sup> using the attenuation data fitted to a 4-th order polynomial exponent formula, since the feature of the buildup factor was changed anomalously and significantly with source energy and penetrations, and essentially dependent upon the group structure of the neutron cross section used.

In this work, the attenuation data for 55 neutron energies above 0.01 MeV up to 400 MeV have been calculated using a high energy neutron group cross section HIL086R library<sup>4)</sup>. The new attenuation factors were also fitted to a polynomial exponent formula modified from the previous one to represent the attenuation in deep penetration. Additional data for finite medium were calculated to estimate the infinite medium effect, as well. These data have been stored in a PKN-H<sup>5)</sup> code, higher energy version of point kernel code PKN<sup>3)</sup>, to apply it for shielding calculations from 0.01 MeV to 400 MeV neutrons.

The attenuation length, which is specially important for shielding calculations of high energy neutrons, has been studied in detail based on the attenuation data, and discussed together with existing calculations and experiments.

## 2. CALCULATIONAL METHOD

### 2.1 Dose Equivalent Calculation

If flux densities for neutrons and secondary gamma rays having energy  $E$  at a location  $r$  are represented by  $\Phi_n(E,r)$  and  $\Phi_\gamma(E,r)$ , the dose equivalents for neutrons and secondary gamma rays are represented, respectively, as follows:

$$H_n(r) = \int [H_n(E)/\Phi_n] \times \Phi_n(E,r) dE , \quad (2.1)$$

$$H_\gamma(r) = \int [H_\gamma(E)/\Phi_\gamma] \times \Phi_\gamma(E,r) dE , \quad (2.2)$$

## 1. INTRODUCTION

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where  $[H_n(E)/\Phi_n]$  and  $[H_\gamma(E)/\Phi_\gamma]$  are the fluence to dose equivalent conversion factor for neutrons and gamma rays.

In the present work, neutron and secondary gamma-ray spectra were calculated for 55 groups of neutron sources from 0.01 MeV up to 400 MeV with ANISN-JR<sup>8)</sup> and HIL086R in the point isotropic geometry of water, ordinary concrete and iron infinite shields. Tables 2.1 and 2.2 show the calculational conditions and the atomic number densities of the materials. For the  $[H_n(E)/\Phi_n]$ , the 1cm dose equivalent conversion factor for the ICRU-sphere phantom presented in Table 21 of ICRP Pub.-51<sup>7)</sup> for neutrons below 20 MeV was used, and the maximum dose equivalent conversion factor for a semi-infinite slab phantom listed in Table 23 of ICRP Pub.-51 for neutrons over 20 MeV up to 400 MeV. For the  $[H_\gamma(E)/\Phi_\gamma]$ , 1cm dose equivalent conversion factor in Table 10 of ICRP Pub.-51 for the ICRU sphere phantom was used for gamma rays less than 10 MeV, and the maximum one in Table 14 of ICRP Pub.-51 for a slab phantom for those greater than 10 MeV.

The dose equivalent conversion factors used are summarized in Tables 2.3 and 2.4 for neutrons and gamma rays, respectively.

## 2.2 Neutron Cross Section Data, HIL086R

The macroscopic effective cross section HIL086R is a revised version of HIL086 neutron cross section<sup>8)</sup>, of which energy group structures of neutrons and gamma-rays are given in Tables 2.5 and 2.6. Among them, 37-neutron groups less than 19.6 MeV and all of gamma-ray groups have been replaced by the data collapsed from JSSTD<sup>9)</sup> group constants for taking into account self-shielding factors. The JSSTD is a group cross section set of 295-neutron and 104-gamma-ray groups based on JENDL-3 cross section library.

In the previous work, a group cross section library JSD100<sup>2)</sup> composed of 100-neutron and 44-gamma-ray groups has been used to calculate the attenuation data for neutrons less than 14.9 MeV. In Figs.2.1 to 2.3, the attenuations for <sup>252</sup>Cf neutron source were compared between HIL086R and JSD100 libraries. The energy group structure of HIL086R is more sparse than that of JSD100, however, the results are in rather good agreement for every material.



### 3. ATTENUATION FACTORS AND FITTING PARAMETERS

#### 3.1 Attenuation Factors of Dose Equivalent

Attenuation factors of neutron and secondary gamma-ray dose equivalents in infinite water, ordinary concrete and iron shields are represented in Tables 3.1 to 3.6 for monoenergetic neutron sources from 0.01 MeV to 400 MeV. In Figs. 3.1 to 3.6, are shown the examples of the attenuation of neutron and secondary gamma-ray dose equivalent for 1st group(400-375 MeV), 5th group(300-275 MeV), 9th group(200-180 MeV), 15th group(100-90 MeV), 22nd group(50-45 MeV) and 29th group(22.5-19.6 MeV) neutron sources. These figures suggest that the attenuations, in general, become smaller with increasing neutron energy and approach an exponential one in deep penetration.

#### 3.2 Fitting Formula for Attenuation Factor

It is useful and convenient to approximate the attenuation factor with a function for the point kernel code PKN-H. Thus, the attenuation data were fitted to a polynomial exponent formula given by the following equations,

$$H_j(r) = \exp( F_j(r) ) / 4\pi r^2, \quad (j = \text{energy index}), \quad (3.1)$$

where

$$F_j(r) = \sum_m a_{mj}^N \times r^{m-1}, \quad (m = 1 \text{ to } N+1), \quad (3.2)$$

$a_{mj}^N$ ;  $m$ -th coefficient of  $N$ -order polynomial for  $j$ -th source energy.

According to the systematic survey in the previous work<sup>1)</sup>,  $N=4$  has been chosen in this study, as well. The parameter set  $a_{mj}^N$  determined by a least square method are summarized in Tables 3.7 to 3.12.

The reproducibility with the parameters determined was examined for 1st(400-375 MeV), 15th(100-90 MeV) and 29th(22.5-19.6 MeV) groups in water, concrete and iron. As seen in Tables 3.13 to 3.15, the fitting error for neutrons is very small except for thickness less than 5-10 cm and the maximum error is almost within 20 % for every source and shield

combination. For secondary gamma-rays, the fitting error becomes fairly larger, especially in the region close to source because of steep increase of secondary gamma-rays, but it doesn't increase monotonously with penetration depth.

### 3.3 Extrapolation of Attenuation Factor

The attenuation data, initially, have been calculated up to the thickness of 14 m for water, 8 m for concrete and 5 m for iron with the ANISN-JR code. Whereas, the shields thicker than these thicknesses have been often employed in high energy accelerators, especially in proton accelerators. Thus, the feature of the attenuation was examined in deep penetration concerning the present attenuation data. Figures 3.7 to 3.9 represent two kinds of attenuation length changing with the penetration, where solid and broken lines show the attenuation length obtained by fitting the attenuation data to an exponential function divided by  $4\pi r^2$  and a simple exponential function. In water and concrete, the attenuation length shown by broken line approaches more quickly to a constant value, while in iron the solid line converges faster to an asymptotic value. These results suggest that the extrapolation is better to be made based on the different fitting formula for water, concrete and iron, as follows:

$$H_j(r) \sim \exp(-r/\lambda_{o^j}), \quad \text{for water and concrete,} \quad (3.3)$$

$$H_j(r) \sim \exp(-r/\lambda_{o^j})/4\pi r^2, \quad \text{for iron,} \quad (3.4)$$

where  $\lambda_{o^j}$  is the attenuation length for energy group of source neutron  $j$ , determined from the attenuation in deeper penetration, where the attenuation length becomes almost constant.

Namely, the attenuation length for water and ordinary concrete, and for iron were calculated by the following relations, respectively. For water and concrete,

$$1/\lambda_{o^j} = -\ln(H_j(r_o)/H_j(r_1))/(r_o-r_1), \quad (3.5)$$

and for iron,

$$1 / \lambda_o^j = - \ln( r_o^2 \times H_j(r_o) / r_1^2 \times H_j(r_1) ) / (r_o - r_1), \quad (3.6)$$

where  $r_o$  and  $r_1$  represent two arbitrary distant points from source, where the dose equivalent attenuates exponentially.

Subsequently, the extrapolated dose equivalent  $H_j(r)$  is given by the following equations, for water and concrete,

$$H_j(r) = H_j(r_{c_j}) \times \exp(-(r-r_{c_j})/\lambda_o^j), \quad (3.7)$$

and for iron,

$$H_j(r) = H_j(r_{c_j}) \times \exp(-(r-r_{c_j})/\lambda_o^j) \times r_{c_j}^2 / r^2, \quad (3.8)$$

where  $H_j(r_{c_j})$  denotes the dose equivalent at the penetration length  $r_{c_j}$  (conjunction radius), which is calculated by Eq.(3.1) with the parameter set  $a_{m_j}^N$  for source neutrons of  $j$ -th energy.

In summary of the above descriptions, the equation including the extrapolation is given by the similar representation as Eq.(3.1), as follows:

$$H_j(r) = \exp( G_j(r) ) / 4\pi r^2, \quad (3.9)$$

where  $G_j(r)$  is given by the followings,

$$G_j(r) = \sum_m a_{m_j}^N \times r^{m-1} \times \theta_+(r) + ( b_j \times r + g(r) ) \times \theta_-(r), \quad (3.10)$$

$$\theta_+(r) = \{ |r_{c_j} - r| - (r_{c_j} - r) \} / 2 \times |r_{c_j} - r|, \quad (3.11)$$

$$\theta_-(r) = 1 - \theta_+(r), \quad (3.12)$$

$$\begin{aligned} g(r) &= \ln(4\pi r^2), & \text{for water and concrete,} \\ &= 0, & \text{for iron,} \end{aligned} \quad (3.13)$$

where  $a_{m_j}^N$ : fitting parameter set ( $m=1 \sim N$ )

$r_{c_j}$ : conjunction radius

$b_j$ : minus of inverse of attenuation length  $\lambda_o^j$

( $b_j = -\rho / \lambda_o^j$ , where  $\rho$  is density).

In this work, the attenuation lengths for the extrapolation have been calculated for 10 MeV to 400 MeV neutron sources using the data of 1000 to 1300 cm for water, 600 to 700 cm for concrete, and 350 to 450 cm for iron, of which the data aren't affected from the outer boundary. In Tables 3.7 to 3.12, the values of  $b_j$  obtained by above procedure are listed together with the parameter set  $a_{m,j}^N$ , where the  $r_{c,j}$  is the conjunction radius for the extrapolation. The attenuation length for each incident neutron energy is summarized in Table 3.16.

### 3.4 Infinite Medium Effect

The present attenuation factors have been calculated in infinite media. Therefore, the dose equivalent obtained using the data becomes overestimation comparing with that for finite ones, which are rather realistic shields. The correction factor, namely "infinite medium effect"  $C_H(E,r)$  is defined as the ratio of dose equivalent  $H(E,r)$  behind finite medium to dose equivalent  $H_\infty(E,r)$  in infinite one at the same penetration  $r$  by the following,

$$C_H(E,r) = H(E,r) / H_\infty(E,r) . \quad (3.14)$$

To estimate the infinite medium effect, the  $H(E,r)$  for finite thickness of water, concrete and iron shields of 5, 15, 30, 50, 100, 200 and 400 cm were calculated for energies for 400, 200, 100, 50, 20, 10, 5, 2, 1, 0.1, 0.01 MeV neutron sources, and the infinite medium effects  $C_H(E,r)$  were obtained using the data as shown in Tables 3.17 to 3.19. Figures 3.10 to 3.15 show the 3-dimensional description of the infinite medium effect vs. source neutron energies of 0.01 to 400 MeV. The figures show that the infinite medium effect changes smoothly with the source neutron energy and the penetration. Besides, the effect becomes constant with increasing the penetration, which makes possible to do the extrapolation vs. the penetration.

The reflection of neutrons in iron is significant, and the infinite medium effect attains to 0.23 at 400 cm radius for neutron source of 400 MeV. However, the dose equivalent  $H(E,r)$  for a shield of finite thickness is estimated without serious excess overestimation using the data of  $C_H(E,r)$  interperated for source energy and extrapolated for shield thickness. The correction procedure has been added to the PKN-H code.

## 4. DISCUSSION OF ATTENUATION LENGTH

So far, the attenuation of dose equivalent for high energy neutron source has been represented as a product of exponential attenuation part and geometrical attenuation part  $1/4\pi r^2$ ,

$$H(r) = H_0 \times \exp(-r/\lambda_{eff}) / 4\pi r^2, \quad (3.15)$$

where  $\lambda_{eff}$  is effective attenuation length, and it has been shown that the value for concrete is independent upon the penetration in the range greater than 200 g/cm<sup>2</sup> thickness.<sup>10)</sup>

As seen in Figs.3.7 to 3.8, however, the attenuation length  $\lambda$  up to 400 MeV neutrons depends on the radius up to fairly deep penetration, and a limited value  $\lambda_0$  of the attenuation length has been determined by Eq.(3.3) for water and concrete. Therefore, Eq.(3.15) have to be written using the attenuation length depending on the penetration radius as a following,

$$H(r) = H_0 \times \exp(-r/\lambda(r)) / 4\pi r^2, \quad (3.16)$$

where  $\lambda(r)$  is equivalent to the effective attenuation length at radius  $r$ . The inverse of  $\lambda(r)$  at distance  $r$  is defined as the differentiation of  $-\ln[r^2H(r)]$  vs  $r$ , and subsequently it is related to  $\lambda_0$  for water and concrete as

$$\lambda(r) = \lambda_0 / (1 - 2\lambda_0/r). \quad (3.17)$$

Eq.(3.17) suggests that the attenuation length  $\lambda(r)$  used in Eq.(3.16) for water and concrete consists with the attenuation length  $\lambda_0$  where the radius is enough large to neglect the  $2\lambda_0/r$ . The attenuation length at  $r = 300, 500, 1000$  cm and infinite distance in water and concrete are compared in Figs.4.1 and 4.2. The figures represent clearly that the attenuation length in water and concrete doesn't become constant for neutron source at least up to 400 MeV. In Fig.4.2, other measured and calculated data are also compared together. As for concrete, Thomas et al.<sup>10)</sup> have recommended the effective attenuation length based on the calculation by Alsmiller et al.<sup>11)</sup> and Wycoff et al.<sup>12)</sup> and the experiment by Gilbert et al.<sup>13)</sup>. Alsmiller et al. calculated the attenuation of dose

equivalent in 15 ft (= 620 cm) thick concrete in plane geometry. Therefore, we have made additional calculations in the same plane geometry to verify the present data, and compared it with the Alsmiller's data. The comparison has been made for 400, 300, 200, 100 and 50 MeV neutrons, and both results were in good agreement despite of a little different atomic composition of concrete as demonstrated for 400 MeV neutrons in Fig.4.3. Thomas et al. has evaluated the attenuation length from the calculations by Alsmiller et al., as shown by a symbol + in Fig.4.2.<sup>10)</sup> As seen in Fig.4.4, however, the attenuation length is still changed significantly in the range calculated by Alsmiller et al. and the values agree almost with the present attenuation length at 500 cm radius calculated in spherical geometry as seen in Fig.4.2.

In Fig.4.5, the attenuation length  $\lambda_0$  in iron are shown with other data, where the broken line shows the result obtained with HIL086 cross section. We have already pointed out that the effect of self-shielding factor in resonance energy region below 1 MeV is serious for the dose attenuation in iron, because the dose equivalent is dominated by neutrons less than 1 MeV in deep penetration even for several hundreds MeV source neutrons.<sup>4)</sup> Figure 4.5 reveals the different cross sections give a difference greater than 50% to the attenuation length. The present attenuation length in iron, which was calculated with HIL086R cross section is constant from 400 MeV down to 1 MeV, and nearly equal to the measurement value for 12 GeV proton incident at KEK<sup>14)</sup>, although the value is larger than the measurements of Howe et al.<sup>15)</sup> and the estimation value of Stevenson et al.<sup>16)</sup>.

## 5. CONCLUDING REMARKS

A lot of attenuation length has been measured and calculated in context with the Moyer model<sup>17)</sup> for neutron shielding in high energy accelerators. In the model, the attenuation of neutrons above 150 MeV is represented with an attenuation length which is independent upon the energy. The present study made clear that this special feature isn't satisfied for neutron sources in the intermediate energy region up to 400 MeV. Shin et al.<sup>18)</sup> have calculated the buildup factor of neutrons in a slab geometry of concrete and iron shields for 12 neutron energies from thermal to 400 MeV based on the multigroup cross section DLC-87/HILO<sup>19)</sup>,

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and proposed to represent the attenuation with the buildup factor and the removal cross section which has been assumed as total cross section in the same sense for gamma-ray source. This idea is physically rigorous, but the values of buildup factor of neutrons are strongly dependent upon the energy group structure of cross section and vary largely with the energy to reflect the complicated feature of cross section. In addition, the absolute values become very large, which have already been indicated by the present author<sup>1)</sup>.

The authors, therefore, have calculated the attenuation data in water, concrete and iron with a point isotropic geometry for 55 neutron energies from 0.01 to 400 MeV based on the revised cross section HIL086R. These attenuation data were fitted to a 4th order polynomial exponent formula having an exponential term for extrapolation to facilitate the application to a point kernel code PKN-H.

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#### REFERENCES

- 1) Kotegawa, H. and Tanaka, S.: "Attenuation Data of Point Isotropic Neutron Sources in the Shielding Materials of Water, Concrete and Iron", JAERI-M 90-174(1990), (in Japanese).
- 2) ORNL-RSIC Data Library Collection, DLC-51/JSD-100/120.
- 3) Kotegawa, H. and Tanaka, S.: "A Point Kernel Shielding Code for Calculations of Neutron and Secondary Gamma Ray 1cm Dose Equivalents: PKN", JAERI-M 91-148(1991), (in Japanese).
- 4) Kotegawa, H. et al.: "Neutron-Photon Multigroup Cross Sections for Neutron Energies up to 400MeV: HIL086R -Revision of HIL086 Library-", JAERI-M 93-020(1993).
- 5) "PKN-H", JAERI-Data/Code 94-xxx(1994), (under preparation).
- 6) Koyama, K. et al.: "ANISN-JR: A One Dimensional Discrete Ordinates Code for Neutron and Gamma-Ray Transport Calculation", JAERI-M 6954(1977).
- 7) ICRP-51: "Data for Use in Protection Against External Radiation", (1987).



and proposed to represent the attenuation with the buildup factor and the removal cross section which has been assumed as total cross section in the same sense for gamma-ray source. This idea is physically rigorous, but the values of buildup factor of neutrons are strongly dependent upon the energy group structure of cross section and vary largely with the energy to reflect the complicated feature of cross section. In addition, the absolute values become very large, which have already been indicated by the present author<sup>1)</sup>.

The authors, therefore, have calculated the attenuation data in water, concrete and iron with a point isotropic geometry for 55 neutron energies from 0.01 to 400 MeV based on the revised cross section HIL086R. These attenuation data were fitted to a 4th order polynomial exponent formula having an exponential term for extrapolation to facilitate the application to a point kernel code PKN-H.

#### ACKNOWLEDGEMENTS

Authors express their gratitude to Shuichi Ban of KEK and Kazuo Shin of Kyoto University for a discussion of attenuation length.

#### REFERENCES

- 1) Kotegawa, H. and Tanaka, S.: "Attenuation Data of Point Isotropic Neutron Sources in the Shielding Materials of Water, Concrete and Iron", JAERI-M 90-174(1990), (in Japanese).
- 2) ORNL-RSIC Data Library Collection, DLC-51/JSD-100/120.
- 3) Kotegawa, H. and Tanaka, S.: "A Point Kernel Shielding Code for Calculations of Neutron and Secondary Gamma Ray 1cm Dose Equivalents: PKN", JAERI-M 91-148(1991), (in Japanese).
- 4) Kotegawa, H. et al.: "Neutron-Photon Multigroup Cross Sections for Neutron Energies up to 400MeV: HIL086R -Revision of HIL086 Library-", JAERI-M 93-020(1993).
- 5) "PKN-H", JAERI-Data/Code 94-xxx(1994), (under preparation).
- 6) Koyama, K. et al.: "ANISN-JR: A One Dimensional Discrete Ordinates Code for Neutron and Gamma-Ray Transport Calculation", JAERI-M 6954(1977).
- 7) ICRP-51: "Data for Use in Protection Against External Radiation", (1987).

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- 4) Kotegawa, H. et al.: "Neutron-Photon Multigroup Cross Sections for Neutron Energies up to 400MeV: HIL086R -Revision of HIL086 Library-", JAERI-M 93-020(1993).
- 5) "PKN-H", JAERI-Data/Code 94-xxx(1994), (under preparation).
- 6) Koyama, K. et al.: "ANISN-JR: A One Dimensional Discrete Ordinates Code for Neutron and Gamma-Ray Transport Calculation", JAERI-M 6954(1977).
- 7) ICRP-51: "Data for Use in Protection Against External Radiation", (1987).

- 8) Alsmiller, R.G., Jr., Barnes, J.M. and Drischler, J.D.: "Neutron-Photon Multigroup Cross Sections for Neutron Energies  $\leq 400$  MeV (Revision 1)", ORNL/TM-9801(1986).
- 9) Hasegawa, A.: "Development of a Common Nuclear Group Constants Library System, JSSTD-295n-104 $\gamma$  Based on JENDL-3 Nuclear Data Library", Nuclear Data Science and Technology, p.232, Springer Verlag(1991).
- 10) Thomas, R.H. and Stevenson, G.R.: "Radiological Safety Aspect of the Operation of Proton Accelerator", Technical Reports Series No.283, IAEA, Vienna(1988).
- 11) Alsmiller, R.G., Jr. et al.: "Shielding Against Neutrons in the Energy Range 50 to 400 MeV", Nucl.Instrum.Methods, 72, 213(1969).
- 12) Wycoff, J.M. and Chilton, A.G.: "Dose Due to Practical Neutron Energy Distributions Incident on Concrete Shielding Walls", Proc. 3rd Int. Conf. IRPA(Snyder W.S., ed.), Rep. CONF-730907-P1, 694(1973).
- 13) Gilbert, W.S. et al.: "1966 CERN-LRL-RHEL Shielding Experiment at the CERN Proton Synchrotron", UCRL-17941(1968).
- 14) Ban, S. et al.: "Measurement of Transverse Attenuation Lengths for Paraffin, Heavy Concrete and Iron around an External Target for 12 GeV Protons", Nucl.Instrum.Methods, 174, 271(1980).
- 15) Howe, H.J., Jr. et al.: "On the Shielding of the External Proton Tunnel Area of Argonne's Zero Gradient Synchrotron", ANL-7273(1966).
- 16) Stevenson, G.R. et al.: "Determination of Transverse Shielding for Proton Accelerator Using the Moyer Model", Health Phys., 43, 13(1982).
- 17) Moyer, B.J.: "Evaluation of Shielding Required for the Improved Bevatron", Rep. UCRL-9769, Lawrence Berkeley Lab., Berkeley(1961).
- 18) Shin, K. and Ishii, Y.: "Build-Up Factors for Medium Energy Neutrons up to 400 MeV", Radiat.Prot.Dosim., 40, 185(1992).
- 19) Alsmiller, R.G., Jr. and Barish, J.: "Neutron-Photon Multigroup Cross Sections for Neutron Energies  $\leq 400$  MeV", ORNL/TM-7818(1981).

Table 2.1 Calculational conditions

Code	ANISN-JR : 1-dimensional discrete ordinates transport calculation code	
Cross section library	HIL086R <sup>*)</sup> : ( P <sub>g</sub> ) effective macroscopic cross section	
Number of Energy Groups	neutron 66 groups gamma-ray 22 groups	
Source	point isotropic 55 monoenergetic neutrons ( from 0.01 to 400MeV )	
Shielding materials	water, ordinary concrete and iron	
Geometry	1-dimensional sphere ( S <sub>10</sub> )	
Flux to dose equivalent conversion factors	neutron	E > 20MeV Maximum dose <sup>**)</sup>
		E ≤ 20MeV 1cm depth dose <sup>***)</sup>
	gamma ray	E > 10MeV Maximum dose <sup>*)</sup>
		E ≤ 10MeV 1cm depth dose <sup>*)</sup>

\*) ICRP Publication 51(1987), p.39, Table 23.

\*\*\*) ibid., p.36, Table 21.

\*) ibid., p.26, Table 14.

\*\*\*) ibid., p.22, Table 10.

Table 2.2 Atomic number densities

unit(10<sup>24</sup>/cm<sup>3</sup>)

	water	ordinary concrete <sup>***)</sup>	iron
H	6.6738-2 <sup>*)</sup>	1.3851-2	
C		1.1542-4	
O	3.3369-2	4.5921-2	
Mg		1.2388-4	
Al		1.7409-3	
Si		1.6621-2	
K		4.6205-4	
Ca		1.5025-3	
Fe		3.4510-4	8.4869-2
density (g/cm <sup>3</sup> )	1.00	2.27	7.87

\*) JAERI-M 6928(1977).

\*\*\*) type 02-a concrete, ANL-5800(1963), p.660.

+) read as 6.6738x10<sup>-2</sup>

Table 2.3 Flux to dose equivalent conversion factors for neutrons

Flux to dose equivalent conversion factors for neutrons			
Neutron energy (MeV)	Maximum dose equiv. ( $10^{-12}\text{Sv}\cdot\text{cm}^2$ )	Neutron energy (MeV)	1cm depth dose equiv. ( $10^{-12}\text{Sv}\cdot\text{cm}^2$ )
700.	1050.	20.	650.
400.	630.	17.	610.
200.	508.	14.	520.
100.	469.	10.	446.
60.	450.	8.	417.
20.	425.	7.	403.
		6.	383.
		5.	378.
		4.	409.
		3.	380.
		2.	352.
		1.5	362.
		1.0	340.
		0.5	258.
		0.2	126.
		0.15	69.
		0.05	35.
		0.02	14.6
		0.01	8.6
		0.001	6.2
		0.0001	7.1
		$1.0\times 10^{-5}$	9.2
		$1.0\times 10^{-6}$	11.2
		$1.0\times 10^{-7}$	10.4
		$2.5\times 10^{-8}$	8.0

\*) ICRP Publication 51(1987), Table 23.  
 \*\*) ibid., Table 21.

Table 2.4 Flux to dose equivalent conversion factors for gamma rays

Flux to dose equivalent conversion factors for gamma rays			
Gamma rays energy (MeV)	Maximum dose equiv. ( $10^{-12}\text{Sv}\cdot\text{cm}^2$ )	Gamma rays energy (MeV)	1cm depth dose equiv. ( $10^{-12}\text{Sv}\cdot\text{cm}^2$ )
20.	41.6	10.	25.18
10.	24.3	8.	21.26
		6.	17.38
		5.	15.43
		4.	13.32
		3.	11.05
		2.	8.475
		1.5	6.916
		1.0	5.096
		0.8	4.280
		0.6	3.380
		0.5	2.880
		0.4	2.381
		0.3	1.808
		0.2	1.181
		0.15	0.893
		0.10	0.612
		0.08	0.531
		0.06	0.503
		0.05	0.527
		0.04	0.614
		0.03	0.786
		0.02	1.010
		0.015	0.846
		0.010	0.0743

\*) ICRP Publication 51(1987), Table 14.  
 \*\*) ibid., Table 10.

Table 2.5 Energy group structure of neutrons in units of MeV

Group	Energy Range	Group	Energy Range
1	4.000E+02 - 3.750E+02	34	1.220E+01 - 1.000E+01
2	3.750E+02 - 3.500E+02	35	1.000E+01 - 8.190E+00
3	3.500E+02 - 3.250E+02	36	8.190E+00 - 6.700E+00
4	3.250E+02 - 3.000E+02	37	6.700E+00 - 5.490E+00
5	3.000E+02 - 2.750E+02	38	5.490E+00 - 4.490E+00
6	2.750E+02 - 2.500E+02	39	4.490E+00 - 3.680E+00
7	2.500E+02 - 2.250E+02	40	3.680E+00 - 3.010E+00
8	2.250E+02 - 2.000E+02	41	3.010E+00 - 2.460E+00
9	2.000E+02 - 1.800E+02	42	2.460E+00 - 2.020E+00
10	1.800E+02 - 1.600E+02	43	2.020E+00 - 1.650E+00
11	1.600E+02 - 1.400E+02	44	1.650E+00 - 1.350E+00
12	1.400E+02 - 1.200E+02	45	1.350E+00 - 1.110E+00
13	1.200E+02 - 1.100E+02	46	1.110E+00 - 9.070E-01
14	1.100E+02 - 1.000E+02	47	9.070E-01 - 7.430E-01
15	1.000E+02 - 9.000E+01	48	7.430E-01 - 4.980E-01
16	9.000E+01 - 8.000E+01	49	4.980E-01 - 3.340E-01
17	8.000E+01 - 7.000E+01	50	3.340E-01 - 2.240E-01
18	7.000E+01 - 6.500E+01	51	2.240E-01 - 1.500E-01
19	6.500E+01 - 6.000E+01	52	1.500E-01 - 8.650E-02
20	6.000E+01 - 5.500E+01	53	8.650E-02 - 3.180E-02
21	5.500E+01 - 5.000E+01	54	3.180E-02 - 1.500E-02
22	5.000E+01 - 4.500E+01	55	1.500E-02 - 7.100E-03
23	4.500E+01 - 4.000E+01	56	7.100E-03 - 3.350E-03
24	4.000E+01 - 3.500E+01	57	3.350E-03 - 1.580E-03
25	3.500E+01 - 3.000E+01	58	1.580E-03 - 4.540E-04
26	3.000E+01 - 2.750E+01	59	4.540E-04 - 1.010E-04
27	2.750E+01 - 2.500E+01	60	1.010E-04 - 2.260E-05
28	2.500E+01 - 2.250E+01	61	2.260E-05 - 1.070E-05
29	2.250E+01 - 1.960E+01	62	1.070E-05 - 5.040E-06
30	1.960E+01 - 1.750E+01	63	5.040E-06 - 2.380E-06
31	1.750E+01 - 1.490E+01	64	2.380E-06 - 1.120E-06
32	1.490E+01 - 1.350E+01	65	1.120E-06 - 4.140E-07
33	1.350E+01 - 1.220E+01	66	4.140E-07 - 1.000E-10

Table 2.6 Energy group structure of gamma rays in units of MeV

Group	Energy Range	Group	Energy Range
1	2.000E+01 - 1.400E+01	12	4.500E+00 - 4.000E+00
2	1.400E+01 - 1.200E+01	13	4.000E+00 - 3.500E+00
3	1.200E+01 - 1.000E+01	14	3.500E+00 - 3.000E+00
4	1.000E+01 - 8.000E+00	15	3.000E+00 - 2.500E+00
5	8.000E+00 - 7.500E+00	16	2.500E+00 - 2.000E+00
6	7.500E+00 - 7.000E+00	17	2.000E+00 - 1.500E+00
7	7.000E+00 - 6.500E+00	18	1.500E+00 - 1.000E+00
8	6.500E+00 - 6.000E+00	19	1.000E+00 - 4.000E-01
9	6.000E+00 - 5.500E+00	20	4.000E-01 - 2.000E-01
10	5.500E+00 - 5.000E+00	21	2.000E-01 - 1.000E-01
11	5.000E+00 - 4.500E+00	22	1.000E-01 - 1.000E-02

Table 3.1 Dose equivalent per source neutrons and attenuation factor of neutrons in water (1/11)

DOSE EQUIVALENT AND ATTENUATION FACTOR		/ WATER		/ TOTAL NEUTRON	
INCIDENT NEUTRON ENERGY(MEV)					
I	R(CM)	1 (400.000 - 375.000)		3 (350.000 - 325.000)	
		H	ATT.F	H	ATT.F
1	5.0	1.85E-08	2.30E+00	1.80E-08	2.30E+00
2	10.0	2.92E-09	1.46E+00	2.85E-09	1.46E+00
3	15.0	1.19E-09	1.33E+00	1.16E-09	1.34E+00
4	20.0	6.47E-10	1.29E+00	6.32E-10	1.29E+00
5	25.0	4.08E-10	1.27E+00	3.98E-10	1.27E+00
6	30.0	2.80E-10	1.26E+00	2.73E-10	1.26E+00
7	35.0	2.04E-10	1.25E+00	1.98E-10	1.24E+00
8	40.0	1.58E-10	1.26E+00	1.54E-10	1.26E+00
9	45.0	1.23E-10	1.24E+00	1.20E-10	1.24E+00
10	50.0	9.86E-11	1.23E+00	9.59E-11	1.23E+00
11	60.0	6.67E-11	1.20E+00	6.48E-11	1.19E+00
12	70.0	4.76E-11	1.16E+00	4.62E-11	1.16E+00
13	80.0	3.53E-11	1.13E+00	3.42E-11	1.12E+00
14	90.0	2.70E-11	1.09E+00	2.61E-11	1.08E+00
15	100.0	2.12E-11	1.06E+00	2.05E-11	1.05E+00
16	110.0	1.68E-11	1.01E+00	1.62E-11	1.00E+00
17	120.0	1.35E-11	9.71E-01	1.30E-11	9.60E-01
18	130.0	1.10E-11	9.28E-01	1.06E-11	9.15E-01
19	140.0	9.05E-12	8.84E-01	8.69E-12	8.71E-01
20	150.0	7.50E-12	8.42E-01	7.19E-12	8.27E-01
21	160.0	6.26E-12	7.99E-01	5.99E-12	7.84E-01
22	170.0	5.26E-12	7.57E-01	5.03E-12	7.43E-01
23	180.0	4.44E-12	7.17E-01	4.24E-12	7.07E-01
24	190.0	3.77E-12	6.78E-01	3.59E-12	6.63E-01
25	200.0	3.21E-12	6.41E-01	3.06E-12	6.25E-01
26	250.0	1.54E-12	4.79E-01	1.45E-12	4.64E-01
27	300.0	7.66E-13	3.44E-01	7.17E-13	3.30E-01
28	350.0	3.94E-13	2.41E-01	3.66E-13	2.29E-01
29	400.0	2.08E-13	1.66E-01	1.92E-13	1.57E-01
30	450.0	1.11E-13	1.17E-01	1.02E-13	1.06E-01
31	500.0	6.05E-14	7.54E-02	5.51E-14	7.05E-02
32	550.0	3.32E-14	5.00E-02	3.01E-14	4.65E-02
33	600.0	1.83E-14	3.29E-02	1.65E-14	3.03E-02
34	650.0	1.02E-14	2.14E-02	9.09E-15	1.96E-02
35	700.0	5.65E-15	1.38E-02	5.03E-15	1.26E-02
36	750.0	3.16E-15	8.86E-03	2.79E-15	8.04E-03
37	800.0	1.77E-15	5.65E-03	1.56E-15	5.10E-03
38	850.0	9.95E-16	3.58E-03	8.71E-16	3.22E-03
39	900.0	5.62E-16	2.27E-03	4.89E-16	2.02E-03
40	950.0	3.17E-16	1.43E-03	2.74E-16	1.26E-03
41	1000.0	1.79E-16	8.90E-04	1.53E-16	7.84E-04
42	1100.0	5.69E-17	3.43E-04	4.84E-17	2.99E-04
43	1200.0	1.82E-17	1.31E-04	1.53E-17	1.13E-04
44	1300.0	5.82E-18	4.90E-05	4.84E-18	4.18E-05

Table 3.1 (Continued) (2/11)

DOSE EQUIVALENT AND ATTENUATION FACTOR		/ WATER		/ TOTAL NEUTRON	
INCIDENT NEUTRON ENERGY(MEV)					
I	R(CM)	6 (275.000 - 250.000)		8 (225.000 - 200.000)	
		H	ATT.F	H	ATT.F
1	5.0	1.60E-08	2.31E+00	1.54E-08	2.32E+00
2	10.0	2.55E-09	1.47E+00	2.46E-09	1.47E+00
3	15.0	1.04E-09	1.35E+00	9.99E-10	1.35E+00
4	20.0	5.62E-10	1.30E+00	5.41E-10	1.30E+00
5	25.0	3.52E-10	1.27E+00	3.39E-10	1.27E+00
6	30.0	2.41E-10	1.25E+00	2.31E-10	1.25E+00
7	35.0	1.75E-10	1.24E+00	1.67E-10	1.23E+00
8	40.0	1.34E-10	1.24E+00	1.29E-10	1.24E+00
9	45.0	1.04E-10	1.22E+00	9.98E-11	1.21E+00
10	50.0	8.29E-11	1.20E+00	7.92E-11	1.19E+00
11	60.0	5.53E-11	1.15E+00	5.28E-11	1.14E+00
12	70.0	3.90E-11	1.11E+00	3.71E-11	1.09E+00
13	80.0	2.86E-11	1.06E+00	2.71E-11	1.04E+00
14	90.0	2.15E-11	1.01E+00	2.03E-11	9.88E-01
15	100.0	1.67E-11	9.64E-01	1.57E-11	9.42E-01
16	110.0	1.31E-11	9.13E-01	1.23E-11	8.90E-01
17	120.0	1.04E-11	8.63E-01	9.70E-12	8.38E-01
18	130.0	8.32E-12	8.13E-01	7.76E-12	7.88E-01
19	140.0	6.75E-12	7.65E-01	6.27E-12	7.38E-01
20	150.0	5.52E-12	7.18E-01	5.11E-12	6.90E-01
21	160.0	4.55E-12	6.73E-01	4.20E-12	6.45E-01
22	170.0	3.77E-12	6.30E-01	3.47E-12	6.01E-01
23	180.0	3.14E-12	5.88E-01	2.88E-12	5.60E-01
24	190.0	2.63E-12	5.49E-01	2.40E-12	5.21E-01
25	200.0	2.21E-12	5.12E-01	2.02E-12	4.84E-01
26	250.0	9.93E-13	3.59E-01	8.88E-13	3.33E-01
27	300.0	4.63E-13	2.41E-01	4.08E-13	2.20E-01
28	350.0	2.24E-13	1.59E-01	1.94E-13	1.43E-01
29	400.0	1.12E-13	1.03E-01	9.49E-14	9.12E-02
30	450.0	5.65E-14	6.61E-02	4.73E-14	5.75E-02
31	500.0	2.90E-14	4.18E-02	2.39E-14	3.59E-02
32	550.0	1.51E-14	2.63E-02	1.22E-14	2.22E-02
33	600.0	7.88E-15	1.64E-02	6.31E-15	1.36E-02
34	650.0	4.15E-15	1.01E-02	3.27E-15	8.29E-03
35	700.0	2.19E-15	6.20E-03	1.70E-15	5.01E-03
36	750.0	1.16E-15	3.79E-03	8.92E-16	3.01E-03
37	800.0	6.21E-16	2.30E-03	4.69E-16	1.80E-03
38	850.0	3.32E-16	1.39E-03	2.48E-16	1.07E-03
39	900.0	1.79E-16	8.38E-04	1.31E-16	6.39E-04
40	950.0	9.60E-17	5.01E-04	6.96E-17	3.77E-04
41	1000.0	5.16E-17	2.98E-04	3.69E-17	2.21E-04
42	1100.0	1.50E-17	1.05E-04	1.04E-17	7.58E-05
43	1200.0	4.36E-18	3.63E-05	2.96E-18	2.56E-05
44	1300.0	1.27E-18	1.24E-05	8.41E-19	8.54E-06















Table 3.2 (Continued) (4/11)

Table with columns for DOSE EQUIVALENT AND ATTENUATION FACTOR, INCIDENT NEUTRON ENERGY (MEV) (16-20), WATER, and SECONDARY GAMMA. Rows 1-44 show data for various R (CM) values from 5.0 to 1300.0.

Table 3.2 (Continued) (5/11)

Table with columns for DOSE EQUIVALENT AND ATTENUATION FACTOR, INCIDENT NEUTRON ENERGY (MEV) (21-25), WATER, and SECONDARY GAMMA. Rows 1-44 show data for various R (CM) values from 5.0 to 1300.0.







Table 3.2 (Continued) (10/11)

Table with columns: DOSE EQUIVALENT AND ATTENUATION FACTOR / WATER / SECONDARY GAMMA, INCIDENT NEUTRON ENERGY(MEV) (46-50), and rows 1-44 for various energies and dose equivalents.

Table 3.2 (Continued) (11/11)

Table with columns: DOSE EQUIVALENT AND ATTENUATION FACTOR / WATER / SECONDARY GAMMA, INCIDENT NEUTRON ENERGY(MEV) (51-55), and rows 1-44 for various energies and dose equivalents.



















Table 3.4 (Continued) (6/11)

DOSE EQUIVALENT	AND ATTENUATION FACTOR		INCIDENT NEUTRON ENERGY(MEV)																																																																																														
			26					27					28					29					30																																																																										
			( 30.000 - 27.500)		( 27.500 - 25.000)		( 25.000 - 22.500)		( 22.500 - 20.000)		( 20.000 - 17.500)		( 17.500 - 15.000)		( 15.000 - 12.500)		( 12.500 - 10.000)		( 10.000 - 7.500)		( 7.500 - 5.000)		( 5.000 - 2.500)		( 2.500 - 0.000)																																																																								
R(CM)	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F																																																																									
1	5.0	1.82E-11	3.62E-03	1.85E-11	3.72E-03	1.92E-11	3.86E-03	1.86E-11	3.78E-03	6.63E-11	1.36E-02	2.46E-11	4.92E-03	1.19E-11	9.69E-03	1.35E-11	2.50E-02	8.48E-12	2.78E-02	5.59E-12	2.86E-02	3.79E-12	2.80E-02	2.59E-12	2.60E-02	1.25E-12	2.07E-02	8.53E-13	1.75E-02	5.82E-13	1.45E-02	3.98E-13	1.17E-02	2.71E-13	9.39E-03	1.84E-13	7.40E-03	1.26E-13	5.79E-03	8.54E-14	4.48E-03	5.79E-14	3.43E-03	3.96E-14	2.63E-03	2.89E-14	1.99E-03	1.82E-14	1.49E-03	8.50E-15	8.44E-04	3.99E-15	4.72E-04	1.89E-15	2.61E-04	8.98E-16	1.44E-04	4.31E-16	7.95E-05	2.08E-16	4.38E-05	1.02E-16	2.41E-05	5.01E-17	1.33E-05	2.49E-17	7.37E-06	1.26E-17	4.14E-06	3.22E-18	1.28E-06	8.48E-19	4.00E-07	3.20E-19	1.76E-07	1.05E-21	1.05E-09	8.24E-22	8.28E-10	4.94E-23	6.43E-11	4.11E-23	5.39E-11	2.18E-24	3.63E-12	1.22E-25	2.50E-13	7.12E-27	1.77E-14	4.32E-28	1.27E-15	2.71E-29	9.39E-17	1.75E-30	7.05E-18	1.17E-31	5.39E-19

Table 3.4 (Continued) (7/11)

DOSE EQUIVALENT	AND ATTENUATION FACTOR		INCIDENT NEUTRON ENERGY(MEV)																																																																																																																								
			31					32					33					34					35																																																																																																				
			( 17.500 - 14.900)		( 14.900 - 13.500)		( 13.500 - 12.200)		( 12.200 - 10.000)		( 10.000 - 8.190)		( 8.190 - 6.380)		( 6.380 - 4.570)		( 4.570 - 2.760)		( 2.760 - 0.950)		( 0.950 - 0.140)		( 0.140 - 0.000)		( 0.000 - 0.000)																																																																																																		
R(CM)	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F	H	ATT.F																																																																																																			
1	5.0	6.35E-11	1.31E-02	8.95E-11	1.87E-02	1.26E-10	2.65E-02	1.15E-10	2.45E-02	9.32E-11	1.99E-02	2.98E-11	2.54E-02	3.14E-11	2.63E-02	4.09E-11	3.44E-02	3.60E-11	3.06E-02	2.98E-11	2.54E-02	1.60E-11	3.36E-02	1.50E-11	2.89E-02	8.80E-12	3.01E-02	5.53E-12	2.95E-02	3.97E-12	3.13E-02	1.11E-11	3.74E-02	9.96E-12	3.38E-02	8.80E-12	3.01E-02	5.89E-12	2.82E-02	6.07E-12	3.22E-02	6.07E-12	3.22E-02	3.88E-12	2.97E-02	3.62E-12	2.79E-02	2.42E-12	2.53E-02	1.67E-12	2.29E-02	1.13E-12	1.96E-02	7.87E-13	1.67E-02	5.20E-13	1.35E-02	3.54E-13	1.09E-02	2.40E-13	8.67E-03	1.63E-13	6.81E-03	1.11E-13	5.32E-03	7.53E-14	4.12E-03	5.10E-14	3.15E-03	3.48E-14	2.41E-03	2.37E-14	1.83E-03	1.61E-14	1.37E-03	7.50E-15	7.76E-04	3.53E-15	4.34E-04	1.67E-15	2.41E-04	7.95E-16	1.33E-04	3.82E-16	7.34E-05	1.85E-16	4.05E-05	9.03E-17	2.23E-05	4.45E-17	1.23E-05	2.21E-17	6.81E-06	1.12E-17	3.83E-06	2.84E-18	1.81E-06	7.43E-19	3.66E-07	1.99E-19	1.15E-07	5.45E-20	3.65E-08	1.52E-20	1.17E-08	6.66E-22	6.97E-10	3.14E-23	4.29E-11	1.56E-24	2.70E-12	8.10E-26	1.73E-13	4.36E-27	1.13E-14	2.43E-28	7.47E-16	1.39E-29	5.02E-17	8.21E-31	3.44E-18	4.97E-32	2.39E-19















Table 3.5 (Continued) (9/11)

Table with columns: DOSE EQUIVALENT AND ATTENUATION FACTOR, IRON, TOTAL NEUTRON, INCIDENT NEUTRON ENERGY(MEV), and rows 1-44 with values for H and ATT.F at various energy levels.

Table 3.5 (Continued) (10/11)

Table with columns: DOSE EQUIVALENT AND ATTENUATION FACTOR, IRON, TOTAL NEUTRON, INCIDENT NEUTRON ENERGY(MEV), and rows 1-44 with values for H and ATT.F at various energy levels.

















Table 3.10 Fitting parameter to dose equivalent of secondary gamma rays in ordinary concrete

H <sub>7</sub> / Secondary Gamma ray/ Up. Bound.	Dose Equivalent in Concrete	Dose Equivalent in Concrete
E(1) = 4.00E+02	-1.773E+01	3.159E-07
E(2) = 3.77E+02	-1.773E+01	3.159E-07
E(3) = 3.50E+02	-1.773E+01	3.159E-07
E(4) = 3.25E+02	-1.773E+01	3.159E-07
E(5) = 3.00E+02	-1.773E+01	3.159E-07
E(6) = 2.75E+02	-1.773E+01	3.159E-07
E(7) = 2.50E+02	-1.773E+01	3.159E-07
E(8) = 2.25E+02	-1.773E+01	3.159E-07
E(9) = 2.00E+02	-1.773E+01	3.159E-07
E(10) = 1.75E+02	-1.773E+01	3.159E-07
E(11) = 1.50E+02	-1.773E+01	3.159E-07
E(12) = 1.25E+02	-1.773E+01	3.159E-07
E(13) = 1.00E+02	-1.773E+01	3.159E-07
E(14) = 7.5E+01	-1.773E+01	3.159E-07
E(15) = 5.0E+01	-1.773E+01	3.159E-07
E(16) = 2.5E+01	-1.773E+01	3.159E-07
E(17) = 1.0E+01	-1.773E+01	3.159E-07
E(18) = 5.0E+00	-1.773E+01	3.159E-07
E(19) = 2.5E+00	-1.773E+01	3.159E-07
E(20) = 1.0E+00	-1.773E+01	3.159E-07
E(21) = 5.0E+00	-1.773E+01	3.159E-07
E(22) = 2.5E+00	-1.773E+01	3.159E-07
E(23) = 1.0E+00	-1.773E+01	3.159E-07
E(24) = 5.0E+00	-1.773E+01	3.159E-07
E(25) = 2.5E+00	-1.773E+01	3.159E-07
E(26) = 1.0E+00	-1.773E+01	3.159E-07
E(27) = 5.0E+00	-1.773E+01	3.159E-07
E(28) = 2.5E+00	-1.773E+01	3.159E-07
E(29) = 1.0E+00	-1.773E+01	3.159E-07
E(30) = 5.0E+00	-1.773E+01	3.159E-07
E(31) = 2.5E+00	-1.773E+01	3.159E-07
E(32) = 1.0E+00	-1.773E+01	3.159E-07
E(33) = 5.0E+00	-1.773E+01	3.159E-07
E(34) = 2.5E+00	-1.773E+01	3.159E-07
E(35) = 1.0E+00	-1.773E+01	3.159E-07
E(36) = 5.0E+00	-1.773E+01	3.159E-07
E(37) = 2.5E+00	-1.773E+01	3.159E-07
E(38) = 1.0E+00	-1.773E+01	3.159E-07
E(39) = 5.0E+00	-1.773E+01	3.159E-07
E(40) = 2.5E+00	-1.773E+01	3.159E-07
E(41) = 1.0E+00	-1.773E+01	3.159E-07
E(42) = 5.0E+00	-1.773E+01	3.159E-07
E(43) = 2.5E+00	-1.773E+01	3.159E-07
E(44) = 1.0E+00	-1.773E+01	3.159E-07
E(45) = 5.0E+00	-1.773E+01	3.159E-07
E(46) = 2.5E+00	-1.773E+01	3.159E-07
E(47) = 1.0E+00	-1.773E+01	3.159E-07
E(48) = 5.0E+00	-1.773E+01	3.159E-07
E(49) = 2.5E+00	-1.773E+01	3.159E-07
E(50) = 1.0E+00	-1.773E+01	3.159E-07
E(51) = 5.0E+00	-1.773E+01	3.159E-07
E(52) = 2.5E+00	-1.773E+01	3.159E-07
E(53) = 1.0E+00	-1.773E+01	3.159E-07
E(54) = 5.0E+00	-1.773E+01	3.159E-07
E(55) = 2.5E+00	-1.773E+01	3.159E-07

Table 3.9 Fitting parameter to dose equivalent of neutrons in ordinary concrete

H <sub>n</sub> / Total Neutron / Up. Bound.	Dose Equivalent in Concrete	Dose Equivalent in Concrete
E(1) = 4.00E+02	-1.227E+01	3.570E-08
E(2) = 3.75E+02	-1.227E+01	3.570E-08
E(3) = 3.50E+02	-1.227E+01	3.570E-08
E(4) = 3.25E+02	-1.227E+01	3.570E-08
E(5) = 3.00E+02	-1.227E+01	3.570E-08
E(6) = 2.75E+02	-1.227E+01	3.570E-08
E(7) = 2.50E+02	-1.227E+01	3.570E-08
E(8) = 2.25E+02	-1.227E+01	3.570E-08
E(9) = 2.00E+02	-1.227E+01	3.570E-08
E(10) = 1.75E+02	-1.227E+01	3.570E-08
E(11) = 1.50E+02	-1.227E+01	3.570E-08
E(12) = 1.25E+02	-1.227E+01	3.570E-08
E(13) = 1.00E+02	-1.227E+01	3.570E-08
E(14) = 7.5E+01	-1.227E+01	3.570E-08
E(15) = 5.0E+01	-1.227E+01	3.570E-08
E(16) = 2.5E+01	-1.227E+01	3.570E-08
E(17) = 1.0E+01	-1.227E+01	3.570E-08
E(18) = 5.0E+00	-1.227E+01	3.570E-08
E(19) = 2.5E+00	-1.227E+01	3.570E-08
E(20) = 1.0E+00	-1.227E+01	3.570E-08
E(21) = 5.0E+00	-1.227E+01	3.570E-08
E(22) = 2.5E+00	-1.227E+01	3.570E-08
E(23) = 1.0E+00	-1.227E+01	3.570E-08
E(24) = 5.0E+00	-1.227E+01	3.570E-08
E(25) = 2.5E+00	-1.227E+01	3.570E-08
E(26) = 1.0E+00	-1.227E+01	3.570E-08
E(27) = 5.0E+00	-1.227E+01	3.570E-08
E(28) = 2.5E+00	-1.227E+01	3.570E-08
E(29) = 1.0E+00	-1.227E+01	3.570E-08
E(30) = 5.0E+00	-1.227E+01	3.570E-08
E(31) = 2.5E+00	-1.227E+01	3.570E-08
E(32) = 1.0E+00	-1.227E+01	3.570E-08
E(33) = 5.0E+00	-1.227E+01	3.570E-08
E(34) = 2.5E+00	-1.227E+01	3.570E-08
E(35) = 1.0E+00	-1.227E+01	3.570E-08
E(36) = 5.0E+00	-1.227E+01	3.570E-08
E(37) = 2.5E+00	-1.227E+01	3.570E-08
E(38) = 1.0E+00	-1.227E+01	3.570E-08
E(39) = 5.0E+00	-1.227E+01	3.570E-08
E(40) = 2.5E+00	-1.227E+01	3.570E-08
E(41) = 1.0E+00	-1.227E+01	3.570E-08
E(42) = 5.0E+00	-1.227E+01	3.570E-08
E(43) = 2.5E+00	-1.227E+01	3.570E-08
E(44) = 1.0E+00	-1.227E+01	3.570E-08
E(45) = 5.0E+00	-1.227E+01	3.570E-08
E(46) = 2.5E+00	-1.227E+01	3.570E-08
E(47) = 1.0E+00	-1.227E+01	3.570E-08
E(48) = 5.0E+00	-1.227E+01	3.570E-08
E(49) = 2.5E+00	-1.227E+01	3.570E-08
E(50) = 1.0E+00	-1.227E+01	3.570E-08
E(51) = 5.0E+00	-1.227E+01	3.570E-08
E(52) = 2.5E+00	-1.227E+01	3.570E-08
E(53) = 1.0E+00	-1.227E+01	3.570E-08
E(54) = 5.0E+00	-1.227E+01	3.570E-08
E(55) = 2.5E+00	-1.227E+01	3.570E-08



Table 3.13 Reproduction ratio of dose equivalent of neutrons and secondary gamma rays in water

Group No.	1		15		29	
	400. ~ 375.		100. ~ 90.		22.5 ~ 19.6	
	n	2nd. γ	n	2nd. γ	n	2nd. γ
5.0	0.642	5.023	0.695	5.313	0.960	6.909
10.0	1.000	1.990	1.035	2.064	1.286	2.490
15.0	1.075	1.266	1.090	1.298	1.238	1.459
20.0	1.095	0.977	1.096	0.992	1.163	1.060
25.0	1.095	0.839	1.085	0.845	1.095	0.862
30.0	1.088	0.763	1.074	0.763	1.039	0.749
35.0	1.079	0.721	1.062	0.716	1.000	0.682
40.0	1.050	0.682	1.027	0.673	0.968	0.642
45.0	1.045	0.675	1.022	0.664	0.932	0.607
50.0	1.040	0.672	1.015	0.657	0.910	0.588
60.0	1.029	0.676	1.009	0.660	0.879	0.581
70.0	1.018	0.696	1.002	0.681	0.874	0.588
80.0	1.010	0.733	0.996	0.709	0.882	0.619
90.0	1.003	0.771	0.993	0.749	0.889	0.657
100.0	0.990	0.808	0.983	0.789	0.894	0.698
120.0	0.984	0.892	0.981	0.882	0.918	0.815
140.0	0.982	0.987	0.982	0.984	0.953	0.955
160.0	0.983	1.071	0.985	1.075	0.998	1.112
180.0	0.984	1.149	0.989	1.158	1.024	1.263
200.0	0.986	1.219	0.994	1.236	1.044	1.400
250.0	0.980	1.314	0.991	1.339	1.079	1.576
300.0	0.989	1.323	1.004	1.378	1.091	1.569
350.0	0.999	1.310	1.012	1.325	1.118	1.431
400.0	1.007	1.218	1.020	1.252	1.137	1.262
450.0	1.011	1.140	1.018	1.142	1.105	1.104
500.0	1.012	1.032	1.016	1.039	1.046	0.978
550.0	1.010	0.964	1.008	0.950	1.012	0.890
600.0	1.008	0.894	1.004	0.893	1.015	0.839
650.0	1.005	0.851	0.997	0.835	0.967	0.805
700.0	1.003	0.830	0.994	0.819	0.939	0.799
750.0	1.001	0.837	0.991	0.820	0.934	0.820
800.0	0.998	0.865	0.989	0.862	0.897	0.856
850.0	0.997	0.907	0.988	0.886	0.890	0.916
900.0	0.992	0.969	0.992	0.986	0.919	0.989
950.0	0.993	1.046	0.999	1.069	0.968	1.087
1000.0	0.996	1.143	1.001	1.141	1.063	1.187
1100.0	1.003	1.297	1.017	1.309	1.125	1.316
1200.0	1.005	1.190	1.014	1.203	1.175	1.199
1300.0	0.999	0.796	0.987	0.785	0.849	0.774

Table 3.14 Reproduction ratio of dose equivalent of neutrons and secondary gamma rays in ordinary concrete

Group No.	1		15		29	
	400. ~ 375.		100. ~ 90.		22.5 ~ 19.6	
	n	2nd. γ	n	2nd. γ	n	2nd. γ
5.0	1.180	4.856	1.162	5.090	1.722	6.039
10.0	1.173	1.880	1.220	1.993	1.425	2.286
15.0	1.092	1.168	1.115	1.216	1.122	1.276
20.0	1.007	0.867	1.037	0.909	1.009	0.942
25.0	0.974	0.743	0.987	0.762	0.936	0.776
30.0	0.953	0.682	0.928	0.665	0.890	0.688
35.0	0.940	0.656	0.915	0.635	0.862	0.641
40.0	0.935	0.652	0.910	0.626	0.847	0.619
45.0	0.934	0.662	0.910	0.632	0.841	0.613
50.0	0.917	0.668	0.914	0.650	0.830	0.611
60.0	0.931	0.727	0.929	0.705	0.848	0.650
70.0	0.947	0.801	0.948	0.780	0.876	0.712
80.0	0.965	0.882	0.962	0.859	0.911	0.789
90.0	0.983	0.963	0.981	0.947	0.948	0.876
100.0	0.999	1.040	0.999	1.032	0.985	0.968
110.0	1.013	1.110	1.015	1.112	1.020	1.061
120.0	1.026	1.170	1.028	1.181	1.051	1.151
130.0	1.036	1.219	1.039	1.238	1.077	1.233
140.0	1.044	1.256	1.049	1.282	1.098	1.305
150.0	1.042	1.270	1.055	1.311	1.095	1.341
160.0	1.047	1.284	1.059	1.327	1.115	1.392
170.0	1.051	1.287	1.062	1.331	1.130	1.427
180.0	1.052	1.280	1.062	1.323	1.140	1.445
190.0	1.052	1.265	1.060	1.305	1.144	1.446
200.0	1.050	1.243	1.047	1.267	1.143	1.432
220.0	1.043	1.183	1.039	1.200	1.129	1.363
240.0	1.032	1.111	1.028	1.119	1.101	1.257
260.0	1.018	1.034	1.013	1.035	1.065	1.133
280.0	1.003	0.961	0.997	0.954	1.019	1.007
300.0	0.979	0.887	0.980	0.881	0.976	0.893
350.0	0.948	0.769	0.946	0.751	0.873	0.679
400.0	0.931	0.723	0.928	0.701	0.835	0.596
450.0	0.936	0.751	0.933	0.730	0.818	0.617
500.0	0.961	0.856	0.960	0.842	0.869	0.753
550.0	1.003	1.033	1.004	1.035	1.017	1.024
600.0	1.047	1.240	1.050	1.265	1.090	1.404
650.0	1.072	1.352	1.075	1.393	1.272	1.631
700.0	1.044	1.192	1.046	1.215	1.131	1.348

Table 3.15 Reproduction ratio of dose equivalent of neutrons and secondary gamma rays in iron  
 Table 3.16 Attenuation lengths ( $\lambda_0$ ) vs. source neutron energies  
 unit (kg/m<sup>2</sup>)

Group No.	1		15		29	
	400. ~ 375.		100. ~ 90.		22.5 ~ 19.6	
En(MeV)	n		n		n	
R (cm)	2nd. $\gamma$		2nd. $\gamma$		2nd. $\gamma$	
5.0	1.855	1.483	1.783	1.358	1.501	0.973
10.0	1.225	0.985	1.214	0.923	1.180	0.807
15.0	0.950	0.872	0.958	0.846	0.972	0.821
20.0	0.855	0.877	0.868	0.881	0.903	0.942
25.0	0.815	0.901	0.829	0.931	0.873	1.073
30.0	0.804	0.927	0.817	0.977	0.864	1.170
35.0	0.810	0.950	0.822	1.009	0.867	1.211
40.0	0.821	0.962	0.831	1.022	0.873	1.198
45.0	0.845	0.976	0.852	1.031	0.888	1.165
50.0	0.875	0.987	0.878	1.033	0.906	1.121
55.0	0.904	0.995	0.905	1.030	0.926	1.076
60.0	0.936	1.002	0.934	1.025	0.947	1.037
65.0	0.967	1.007	0.963	1.019	0.968	1.004
70.0	0.998	1.012	0.992	1.013	0.989	0.978
75.0	1.027	1.016	1.018	1.008	1.009	0.958
80.0	1.053	1.020	1.044	1.004	1.028	0.943
85.0	1.077	1.024	1.067	1.001	1.045	0.933
90.0	1.098	1.027	1.087	0.998	1.061	0.927
95.0	1.115	1.029	1.104	0.997	1.075	0.924
100.0	1.118	1.022	1.107	0.987	1.076	0.914
110.0	1.136	1.026	1.128	0.989	1.094	0.921
120.0	1.140	1.028	1.135	0.993	1.103	0.933
130.0	1.133	1.028	1.132	0.997	1.102	0.949
140.0	1.116	1.027	1.118	1.001	1.094	0.967
150.0	1.092	1.023	1.097	1.004	1.078	0.985
160.0	1.063	1.018	1.070	1.007	1.057	1.003
170.0	1.031	1.011	1.040	1.009	1.033	1.018
180.0	0.999	1.003	1.008	1.009	1.008	1.031
190.0	0.969	0.995	0.977	1.008	0.983	1.042
200.0	0.938	0.984	0.944	1.003	0.956	1.046
220.0	0.894	0.970	0.897	0.997	0.918	1.053
240.0	0.871	0.961	0.870	0.991	0.897	1.048
260.0	0.872	0.959	0.868	0.987	0.896	1.034
280.0	0.897	0.966	0.891	0.985	0.916	1.015
300.0	0.948	0.981	0.941	0.988	0.956	0.994
350.0	1.148	1.040	1.147	1.007	1.113	0.946
400.0	1.263	1.079	1.272	1.031	1.199	0.949
450.0	0.870	0.957	0.868	0.985	0.898	1.040

Group	En(MeV)	Attenuation Lengths( $\lambda_0$ )		
		Water	Concrete	Iron
1	400 - 375	880	1030	1820
2	375 - 350	870	1020	1820
3	350 - 325	860	1010	1820
4	325 - 300	840	1000	1820
5	300 - 275	830	990	1820
6	275 - 250	810	980	1820
7	250 - 225	790	960	1820
8	225 - 200	770	950	1820
9	200 - 180	750	930	1820
10	180 - 160	720	890	1820
11	160 - 140	690	860	1820
12	140 - 120	640	820	1820
13	120 - 110	590	770	1820
14	110 - 100	560	740	1820
15	100 - 90	530	700	1820
16	90 - 80	490	660	1820
17	80 - 70	450	610	1820
18	70 - 65	400	560	1820
19	65 - 60	380	530	1820
20	60 - 55	350	500	1820
21	55 - 50	330	470	1820
22	50 - 45	310	440	1820
23	45 - 40	280	420	1820
24	40 - 35	260	390	1820
25	35 - 30	230	360	1820
26	30 - 27.5	190	320	1820
27	27.5 - 25.0	180	310	1820
28	25.0 - 22.5	160	300	1820
29	22.5 - 20.0	140	290	1820
30	20.0 - 19.6	120	270	1820
31	19.6 - 17.5	120	250	1820
32	17.5 - 14.9	120	250	1820

Table 3.17 Infinite medium effect  $C_H(E_n, r)$  of neutrons and secondary gamma rays for water (1/2)

neutron							
MeV cm	400.	200.	100.	50.	20.	10.	
5.	0.99	0.98	0.97	0.96	0.93	0.90	
15.	0.97	0.95	0.94	0.92	0.87	0.83	
30.	0.95	0.94	0.92	0.90	0.85	0.80	
50.	0.94	0.92	0.91	0.89	0.83	0.79	
100.	0.92	0.91	0.90	0.88	0.81	0.78	
200.	0.91	0.90	0.89	0.86	0.81	0.78	
400.	0.89	0.89	0.88	0.85	0.80		

Table 3.17 Infinite medium effect  $C_H(E_n, r)$  of neutrons and secondary gamma rays for water (2/2)

neutron							
MeV cm	5.0	2.0	1.0	0.1	0.01		
5.	0.88	0.81	0.74	0.26	0.20		
15.	0.79	0.64	0.41	0.16	0.16		
30.	0.76	0.57	0.27	0.16	0.16		
50.	0.75	0.56	0.21	0.20	0.24		
100.	0.74	0.56	0.20	0.27	0.27		
200.	0.74	0.55	0.20	0.29	0.29		
400.	0.74	0.55	0.20				

Secondary gamma ray

MeV cm	400.	200.	100.	50.	20.	10.
5.	0.05	0.05	0.06	0.06	0.09	0.38
15.	0.22	0.23	0.24	0.26	0.31	0.48
30.	0.35	0.37	0.40	0.43	0.52	0.68
50.	0.41	0.43	0.47	0.52	0.65	0.80
100.	0.41	0.44	0.49	0.58	0.78	0.85
200.	0.38	0.41	0.47	0.60	0.86	0.87
400.	0.37	0.39	0.45	0.67	0.91	

Secondary gamma ray

MeV cm	5.0	2.0	1.0	0.1	0.01
5.	0.04	0.06	0.09	0.19	0.25
15.	0.41	0.58	0.69	0.81	0.84
30.	0.72	0.85	0.87	0.87	0.87
50.	0.82	0.84	0.84	0.84	0.84
100.	0.81	0.81	0.81	0.81	0.81
200.	0.80	0.80	0.81	0.80	0.80
400.	0.80	0.80	0.81		

Table 3.18 Infinite medium effect  $C_H(E_n, r)$  of neutrons and secondary gamma rays for ordinary concrete (1/2)

neutron								
MeV cm	400.	200.	100.	50.	20.	10.		
5.	0.93	0.91	0.89	0.87	0.83	0.81		
15.	0.80	0.77	0.74	0.72	0.68	0.66		
30.	0.74	0.71	0.69	0.67	0.62	0.60		
50.	0.71	0.69	0.67	0.64	0.59	0.57		
100.	0.69	0.67	0.66	0.63	0.57	0.55		
200.	0.67	0.66	0.64	0.62	0.56	0.53		
400.	0.65	0.65	0.64	0.61	0.55	0.53		

Table 3.18 Infinite medium effect  $C_H(E_n, r)$  of neutrons and secondary gamma rays for ordinary concrete (2/2)

neutron						
MeV cm	5.	2.	1.	0.1	0.01	
5.	0.77	0.70	0.66	0.63	0.38	
15.	0.61	0.57	0.56	0.31	0.26	
30.	0.57	0.51	0.38	0.25	0.24	
50.	0.54	0.41	0.26	0.24	0.24	
100.	0.50	0.27	0.24	0.24	0.24	
200.	0.47	0.24	0.24	0.27	0.26	
400.	0.46	0.24	0.27	0.29	0.27	

Secondary gamma ray

MeV cm	400.	200.	100.	50.	20.	10.
5.	0.13	0.13	0.15	0.15	0.21	0.37
15.	0.15	0.15	0.16	0.17	0.23	0.23
30.	0.22	0.24	0.26	0.29	0.35	0.32
50.	0.30	0.32	0.36	0.41	0.50	0.50
100.	0.33	0.36	0.41	0.51	0.68	0.76
200.	0.32	0.35	0.40	0.54	0.86	0.92
400.	0.32	0.35	0.40	0.56	0.93	0.94

Secondary gamma ray

MeV cm	5.	2.	1.	0.1	0.01
5.	0.37	0.05	0.01	0.01	0.04
15.	0.23	0.09	0.12	0.20	0.30
30.	0.32	0.37	0.45	0.53	0.60
50.	0.50	0.63	0.70	0.75	0.78
100.	0.76	0.88	0.90	0.91	0.91
200.	0.93	0.94	0.94	0.94	0.94
400.	0.94	0.94	0.94	0.94	0.94

Table 3.19 Infinite medium effect  $C_H(E_n, r)$  of neutrons and secondary gamma rays for iron (1/2)

neutron										
MeV cm	400.	200.	100.	50.	20.	10.				
5.	0.54	0.55	0.57	0.58	0.58	0.61				
15.	0.34	0.35	0.36	0.37	0.38	0.38				
30.	0.30	0.30	0.31	0.32	0.32	0.32				
50.	0.29	0.29	0.30	0.30	0.29	0.29				
100.	0.28	0.28	0.28	0.27	0.27	0.27				
200.	0.26	0.25	0.24	0.23	0.23	0.23				
400.	0.23	0.23	0.22	0.22	0.22	0.22				

Table 3.19 Infinite medium effect  $C_H(E_n, r)$  of neutrons and secondary gamma rays for iron (2/2)

neutron										
MeV cm	5.	2.	1.	0.1	0.01					
5.	0.57	0.53	0.59	0.48	0.23					
15.	0.38	0.38	0.39	0.33	0.16					
30.	0.31	0.31	0.32	0.27	0.17					
50.	0.29	0.29	0.29	0.23	0.19					
100.	0.27	0.27	0.26	0.22	0.20					
200.	0.24	0.23	0.23	0.22	0.21					
400.	0.23	0.22	0.22	0.22	0.21					

Secondary gamma ray

Secondary gamma ray										
MeV cm	400.	200.	100.	50.	20.	10.				
5.	0.36	0.37	0.39	0.40	0.39	0.67				
15.	0.33	0.34	0.36	0.38	0.42	0.59				
30.	0.25	0.25	0.25	0.25	0.24	0.30				
50.	0.17	0.16	0.15	0.13	0.11	0.10				
100.	0.14	0.13	0.13	0.13	0.13	0.13				
200.	0.15	0.15	0.16	0.16	0.16	0.16				
400.	0.16	0.16	0.16	0.16	0.16	0.16				

Secondary gamma ray

Secondary gamma ray										
MeV cm	5.	2.	1.	0.1	0.01					
5.	0.59	0.45	0.45	0.19	0.13					
15.	0.51	0.30	0.23	0.08	0.14					
30.	0.19	0.12	0.10	0.09	0.29					
50.	0.10	0.09	0.10	0.13	0.40					
100.	0.13	0.13	0.14	0.16	0.52					
200.	0.15	0.16	0.16	0.16	0.56					
400.	0.16	0.16	0.16	0.16	0.57					

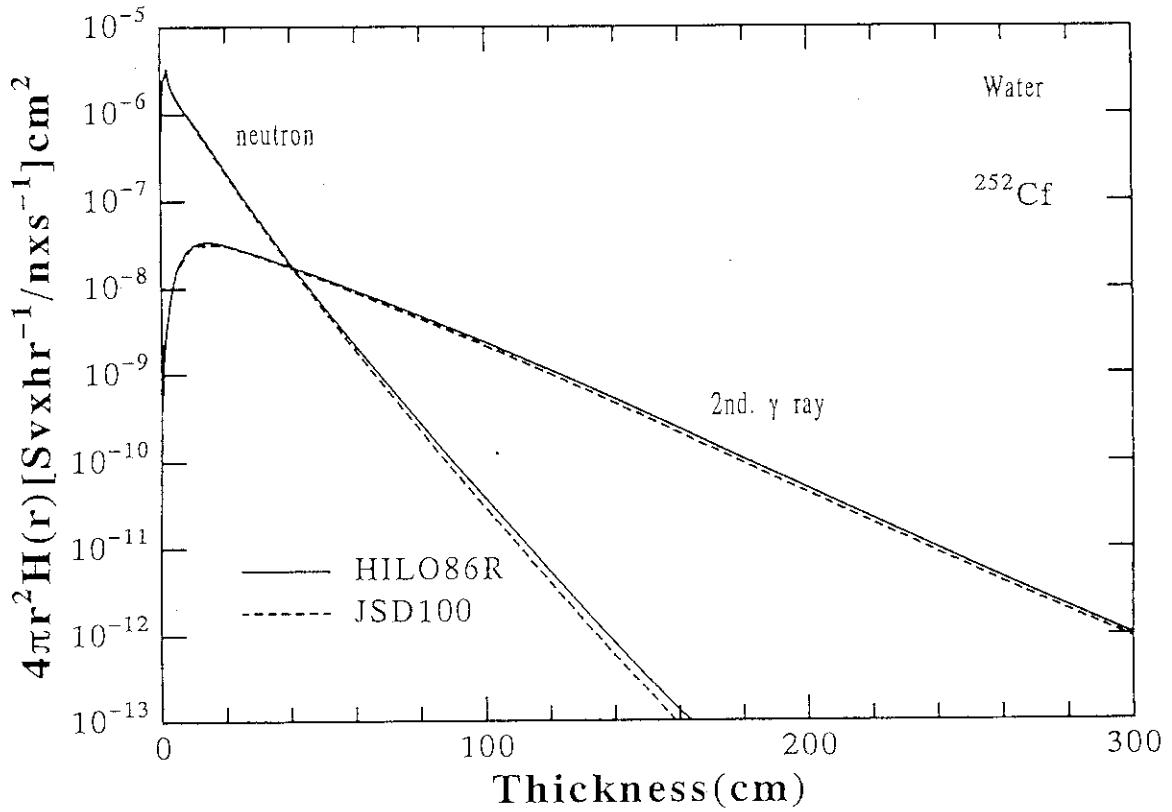


Fig.2.1 Comparison of dose equivalent in water calculated with HILO86R and JSD100 for  $^{252}\text{Cf}$  neutron source.

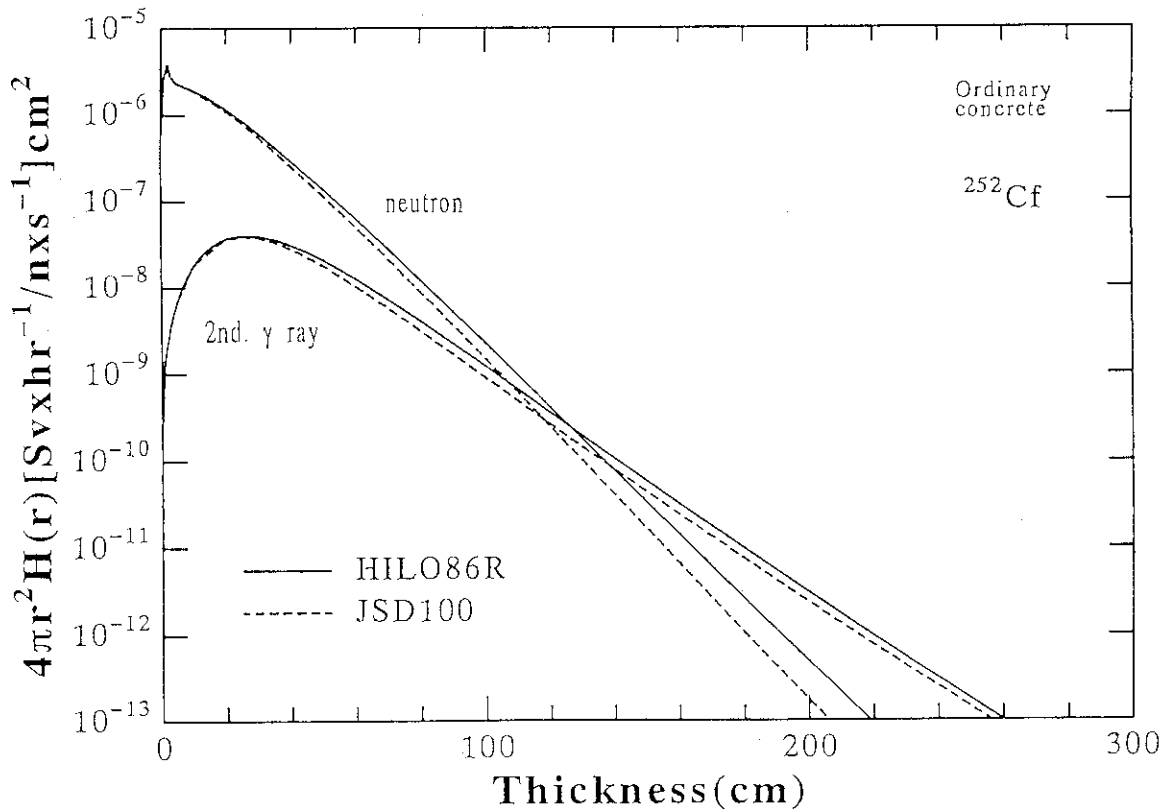


Fig.2.2 Comparison of dose equivalent in ordinary concrete calculated with HILO86R and JSD100 for  $^{252}\text{Cf}$  neutron source.



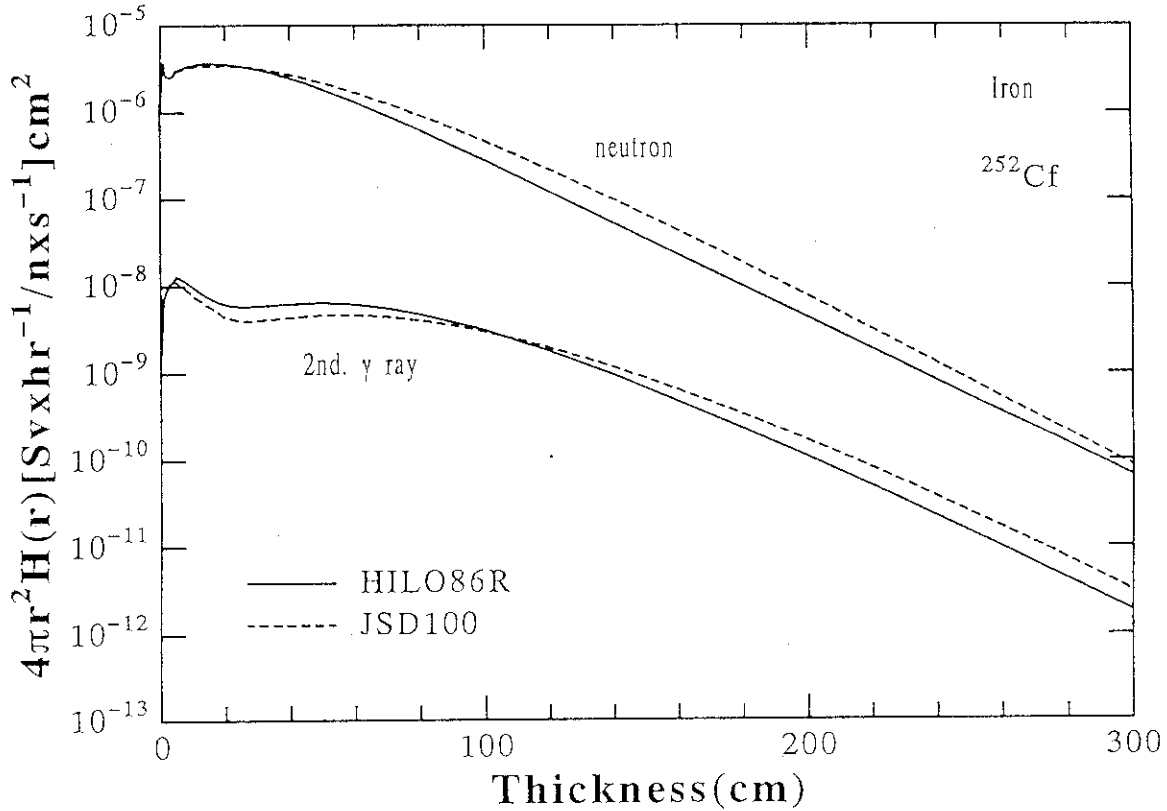


Fig.2.3 Comparison of dose equivalent in iron calculated with HILO86R and JSD100 for  $^{252}\text{Cf}$  neutron source.

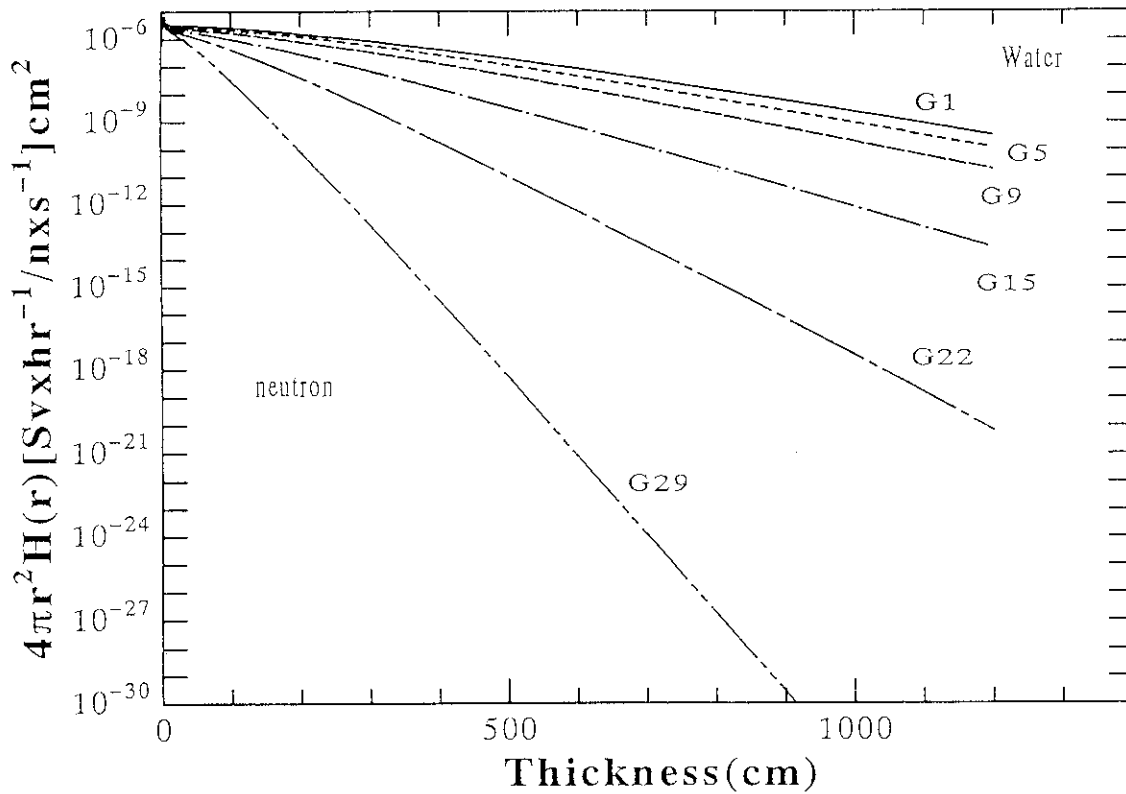


Fig.3.1 Neutron dose equivalent in water for monoenergetic neutron sources (G1=400.-375., G5=300.-275., G9=200.-180., G15=100.-90., G22=50.-45., G29=22.5-19.6MeV).

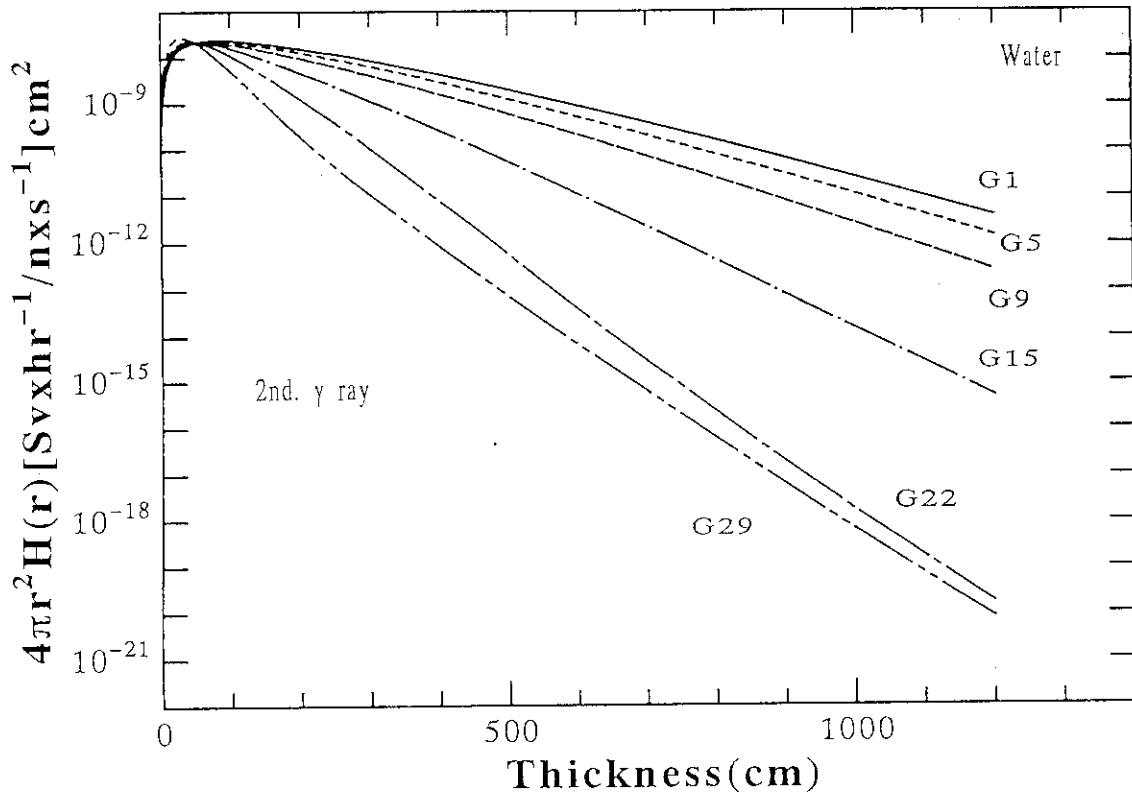


Fig.3.2 Secondary gamma-ray dose equivalent in water for monoenergetic neutron sources (G1=400.-375., G5=300.-275., G9=200.-180., G15=100.-90., G22=50.-45., G29=22.5-19.6MeV).

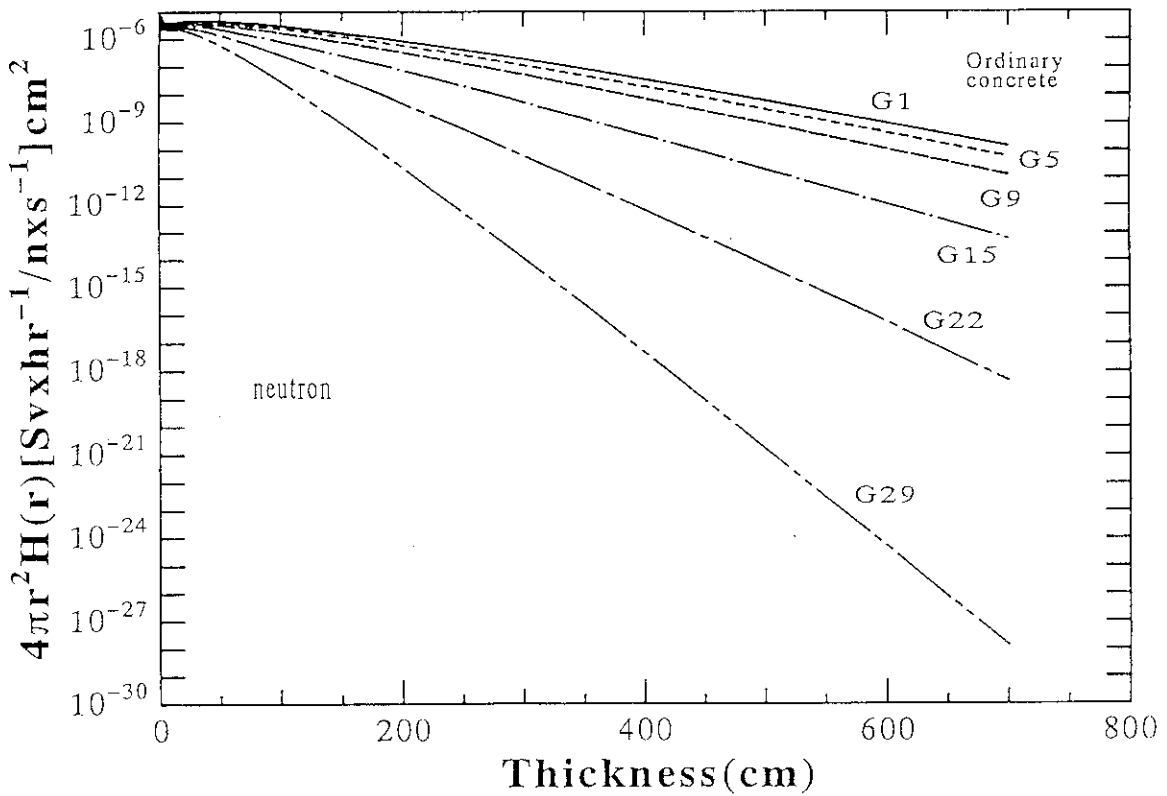


Fig.3.3 Neutron dose equivalent in ordinary concrete for monoenergetic neutron sources (G1=400.-375., G5=300.-275., G9=200.-180., G15=100.-90., G22=50.-45., G29=22.5-19.6MeV).

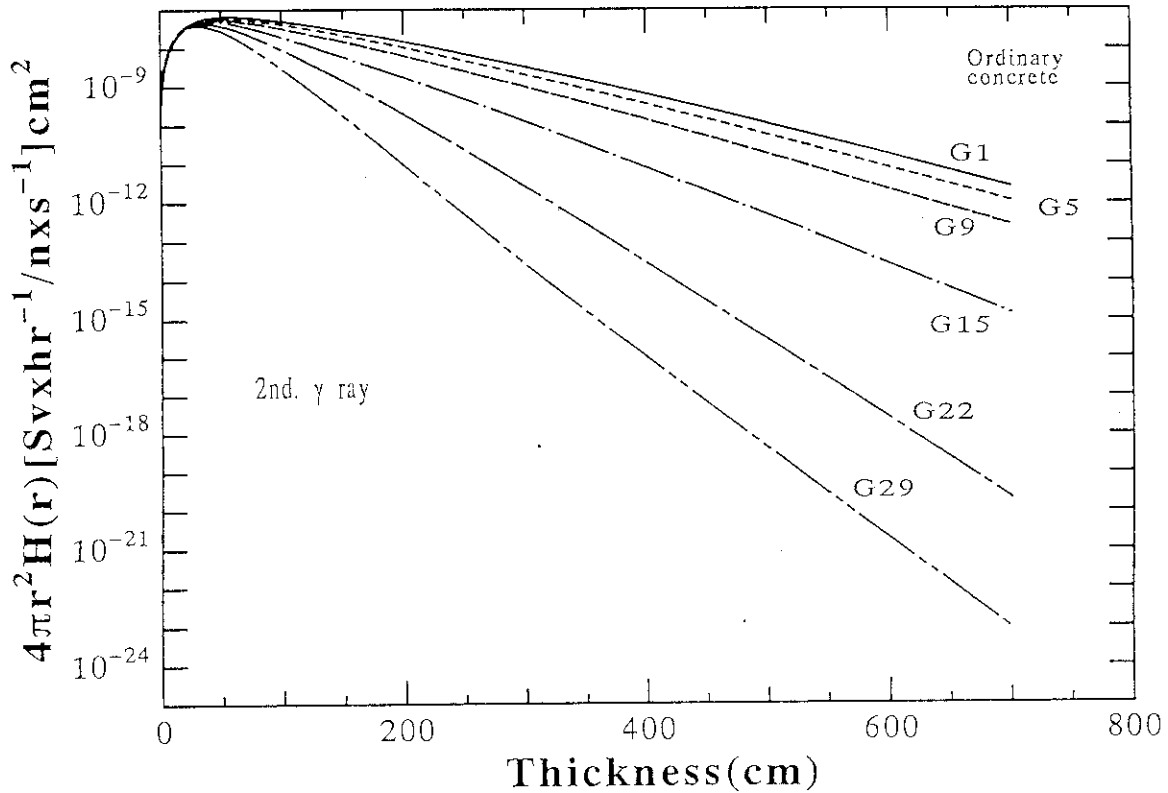


Fig.3.4 Secondary gamma-ray dose equivalent in ordinary concrete for monoenergetic neutron sources (G1=400.-375., G5=300.-275., G9=200.-180., G15=100.-90., G22=50.-45., G29=22.5-19.6MeV).

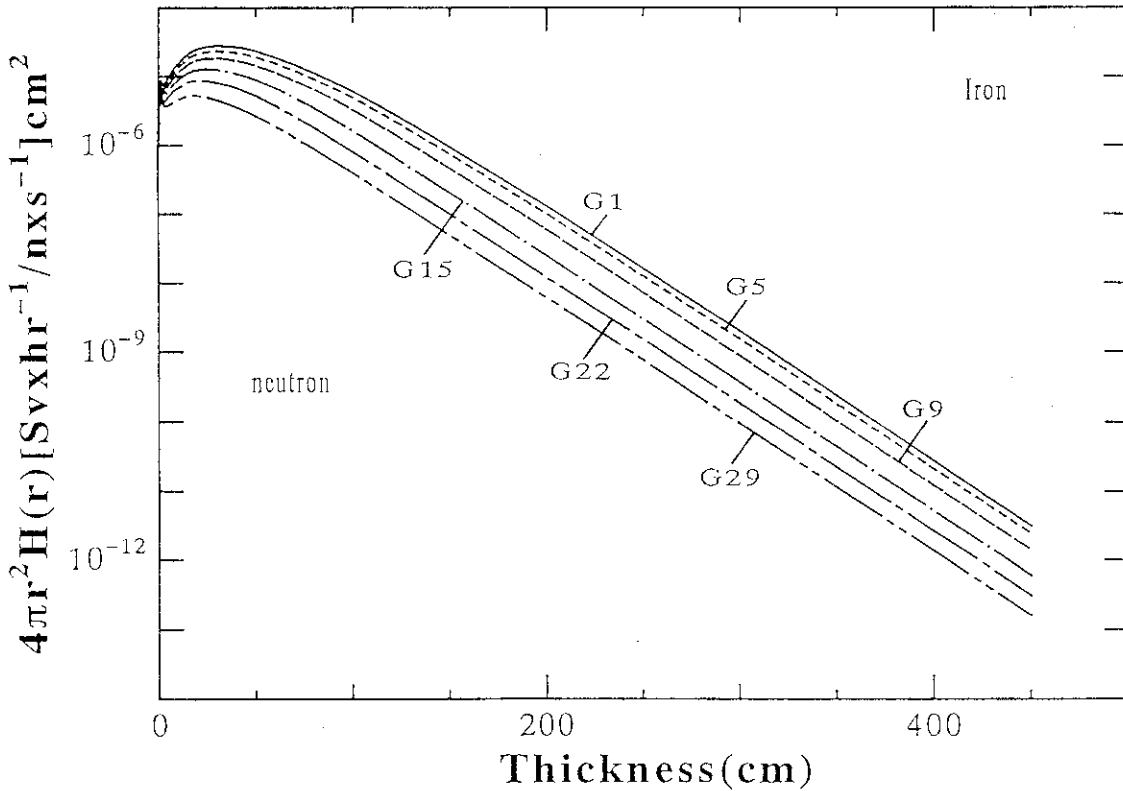


Fig.3.5 Neutron dose equivalent in iron for monoenergetic neutron sources (G1=400.-375., G5=300.-275., G9=200.-180., G15=100.-90., G22=50.-45., G29=22.5-19.6MeV)

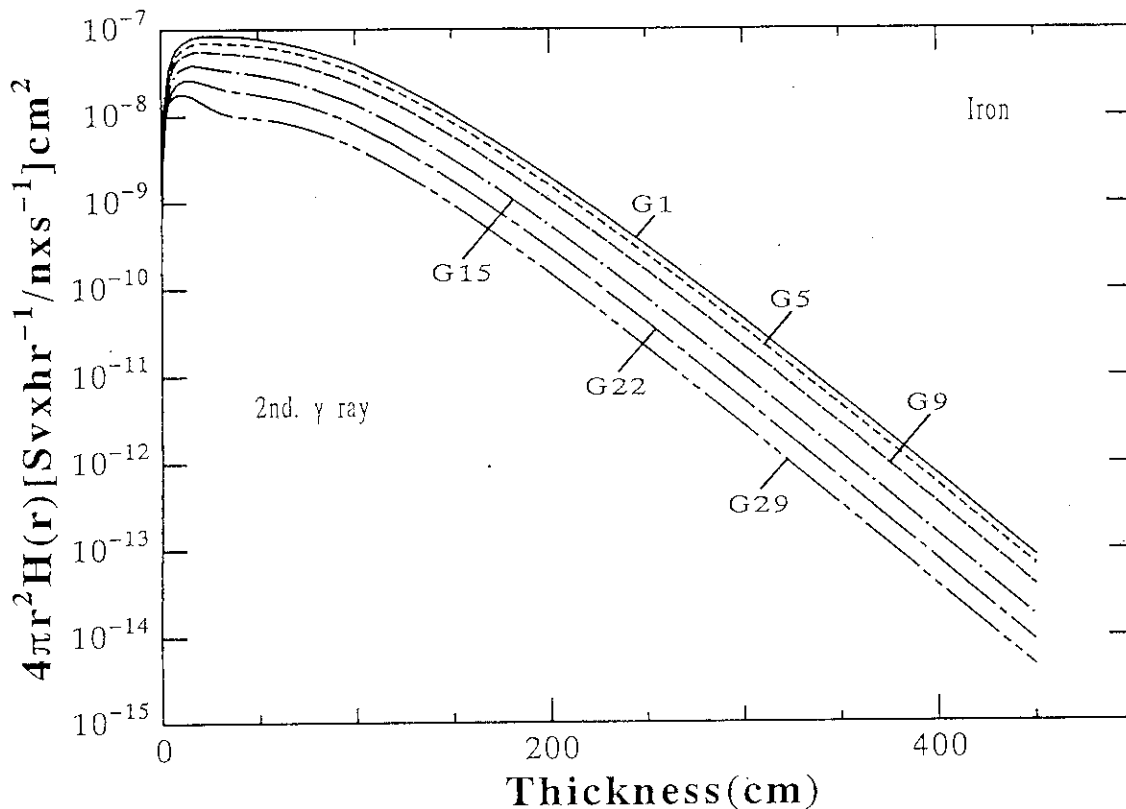


Fig.3.6 Secondary gamma-ray dose equivalent in iron for monoenergetic neutron sources (G1=400.-375., G5=300.-275., G9=200.-180., G15=100.-90., G22=50.-45., G29=22.5-19.6MeV).

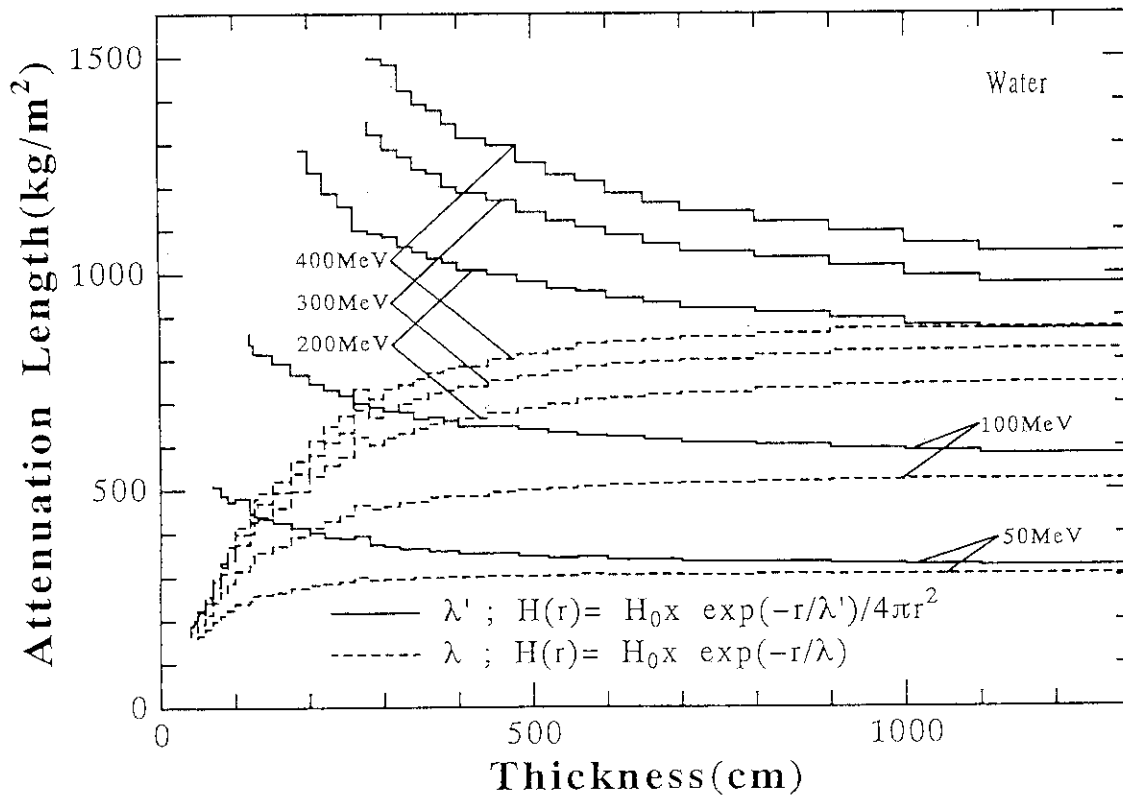


Fig.3.7 Attenuation length vs. thickness for different sources in water.

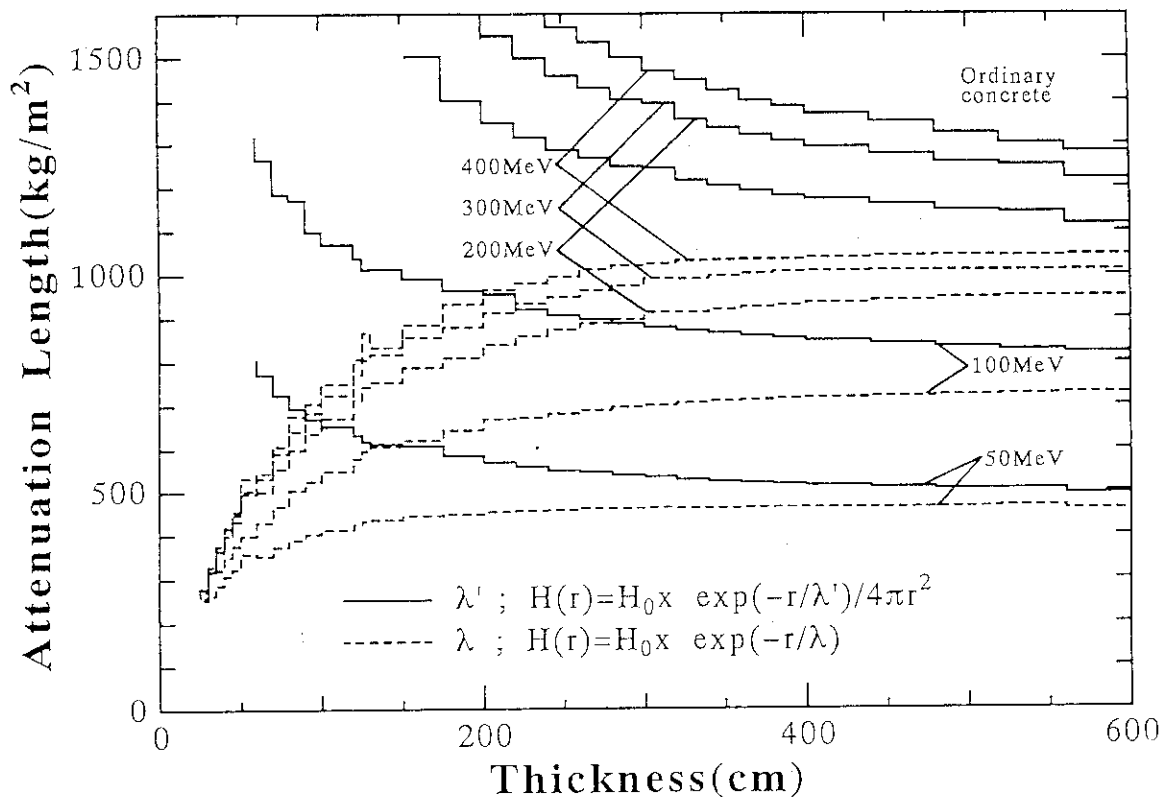


Fig.3.8 Attenuation length vs. thickness for different sources in ordinary concrete.

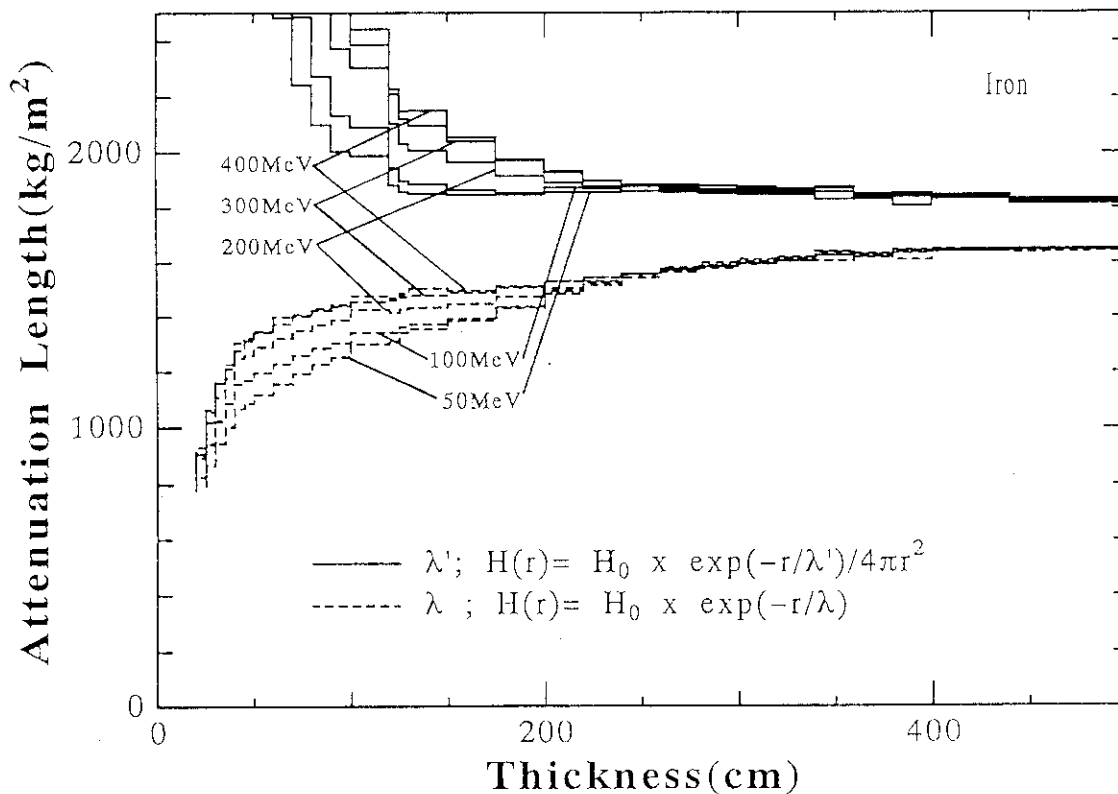


Fig.3.9 Attenuation length vs. thickness for different sources in iron.

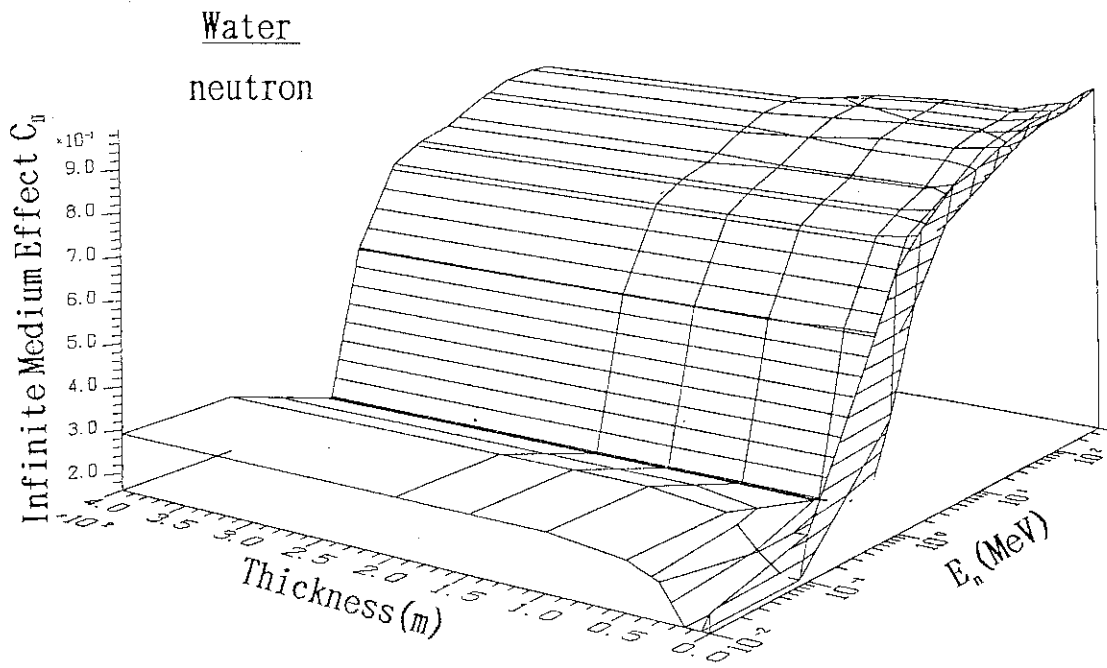


Fig.3.10 Infinite medium effect  $C_H(E_n, r)$  of dose equivalent for neutrons in water.

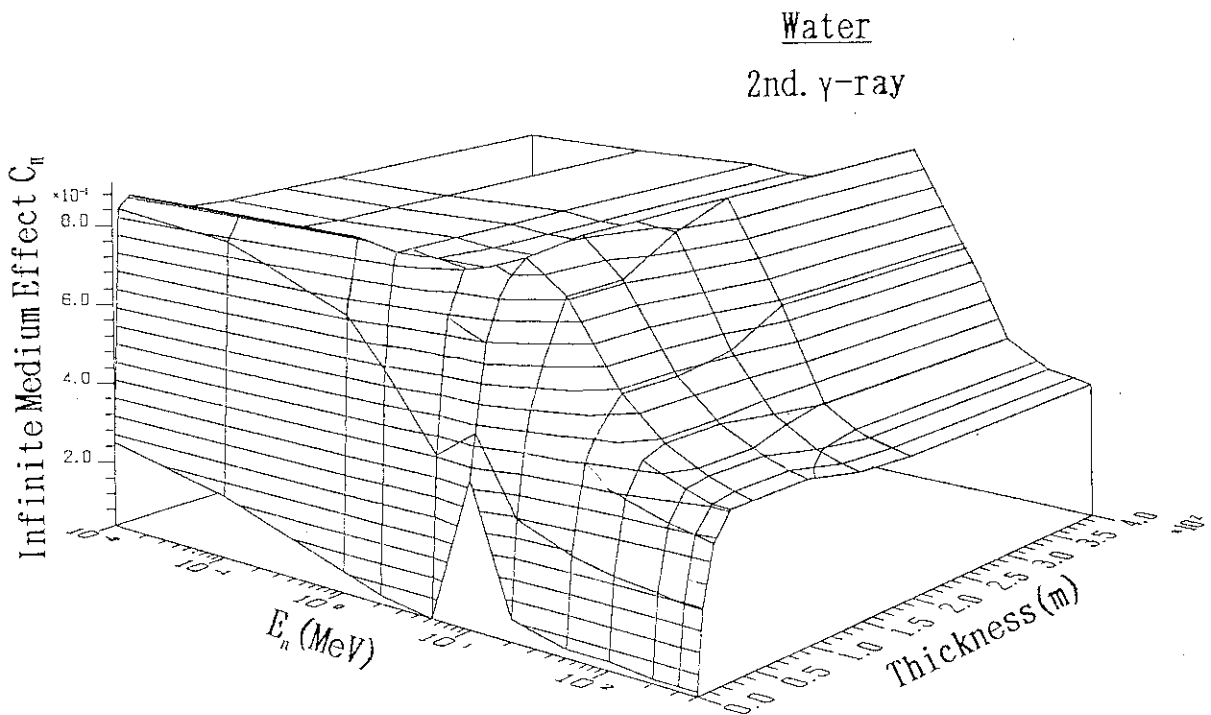


Fig.3.11 Infinite medium effect  $C_H(E_n, r)$  of dose equivalent for secondary gamma rays in water.

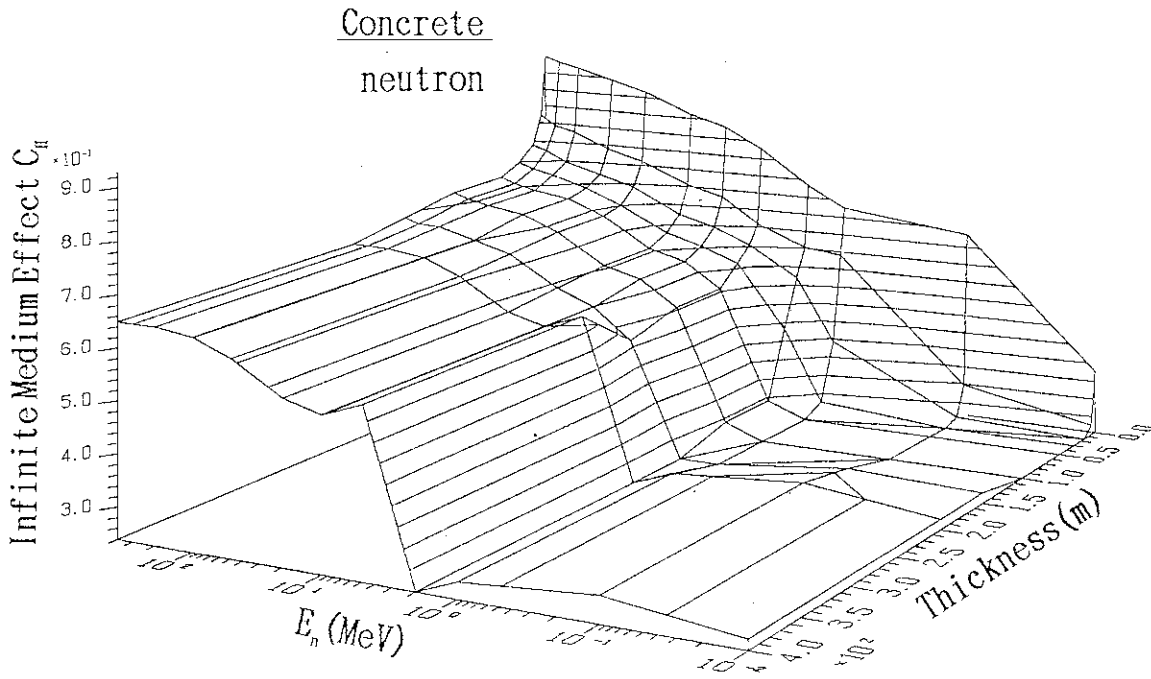


Fig.3.12 Infinite medium effect  $C_H(E_n, r)$  of dose equivalent for neutrons in ordinary concrete.

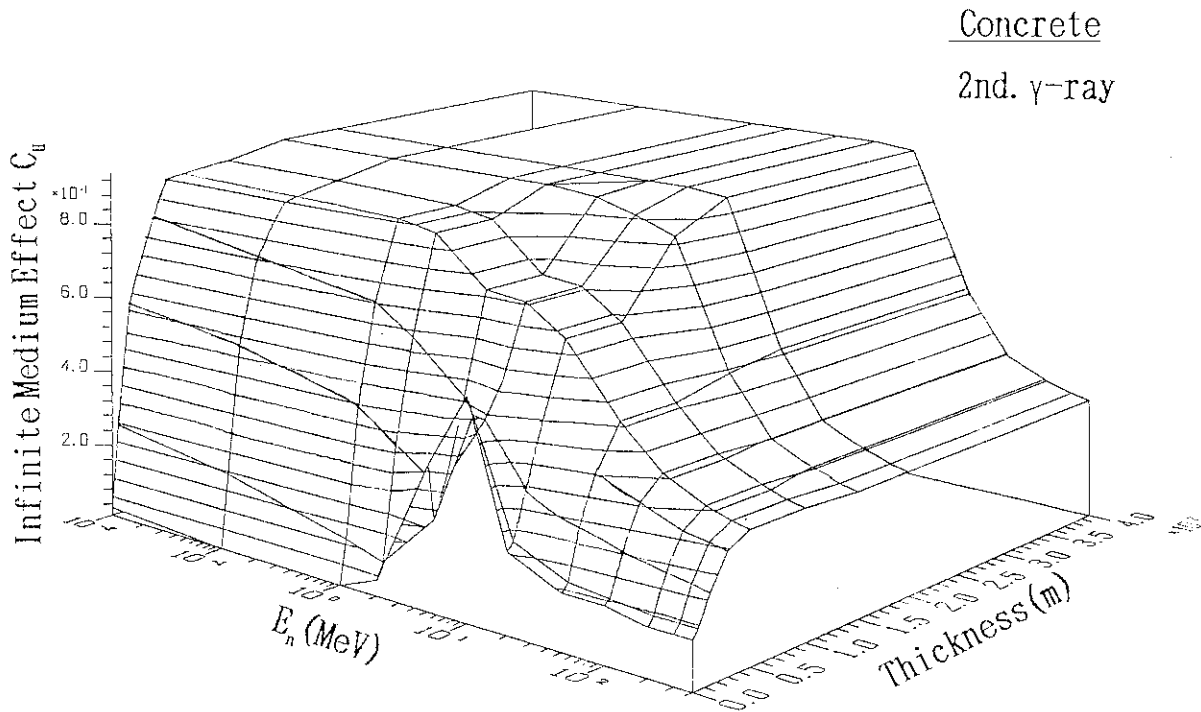


Fig.3.13 Infinite medium effect  $C_H(E_n, r)$  of dose equivalent for secondary gamma rays in ordinary concrete.

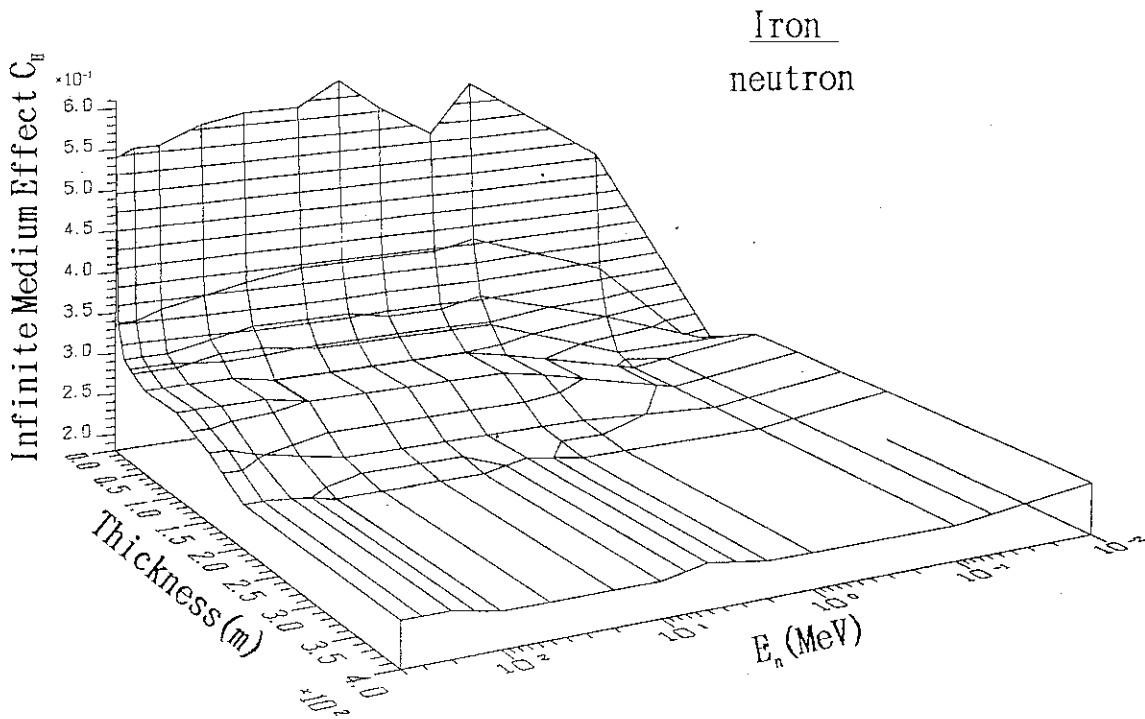


Fig.3.14 Infinite medium effect  $C_H(E_n, r)$  of dose equivalent for neutrons in iron.

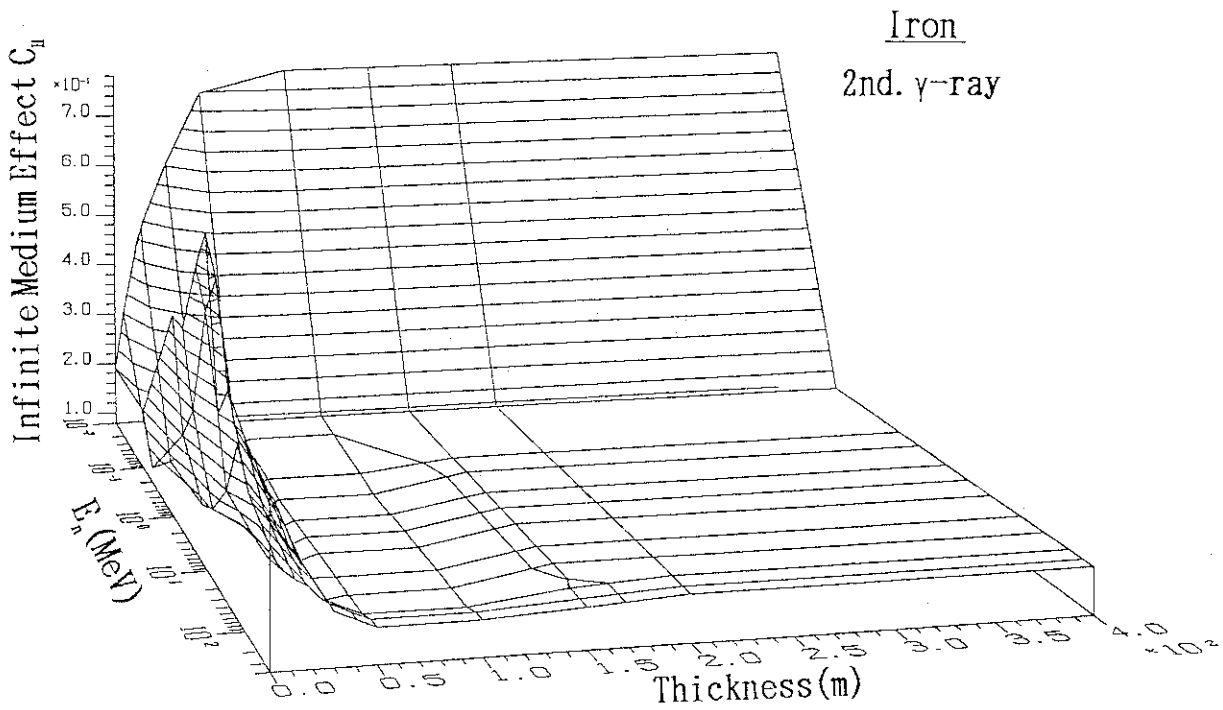


Fig.3.15 Infinite medium effect  $C_H(E_n, r)$  of dose equivalent for secondary gamma rays in iron.



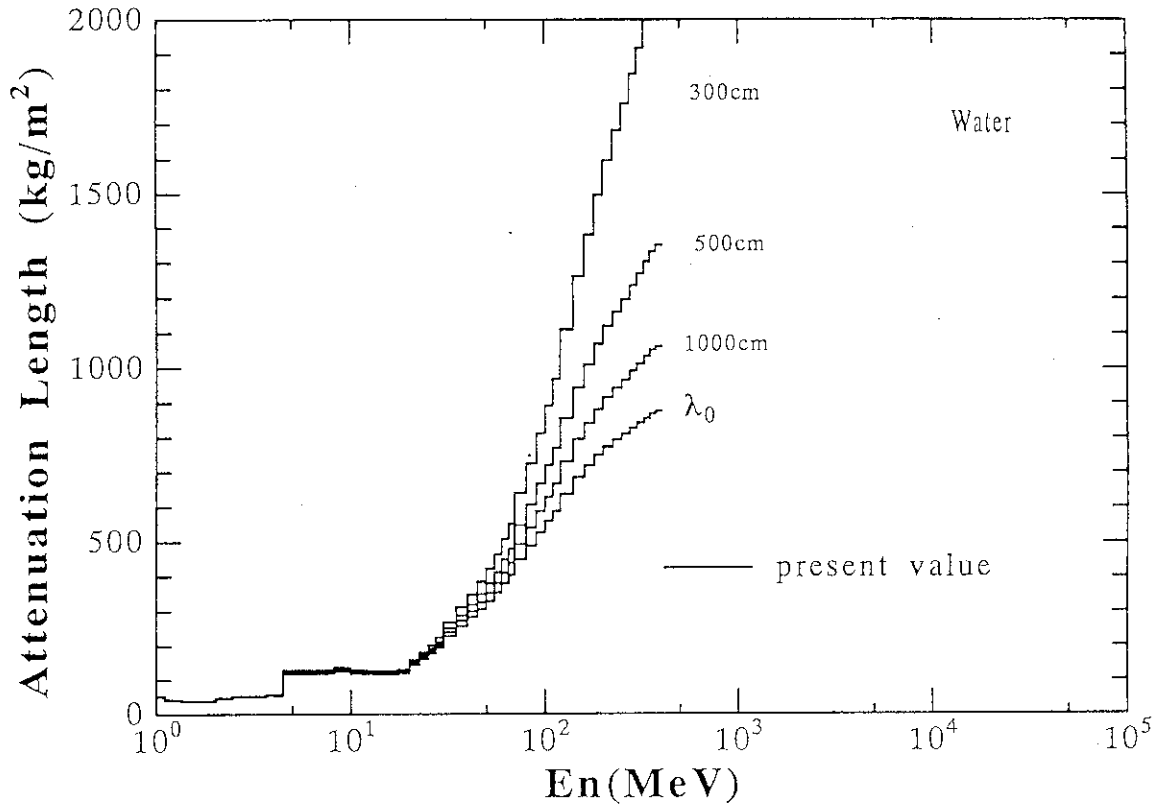


Fig.4.1 Energy dependence of attenuation length obtained at different thicknesses in water.

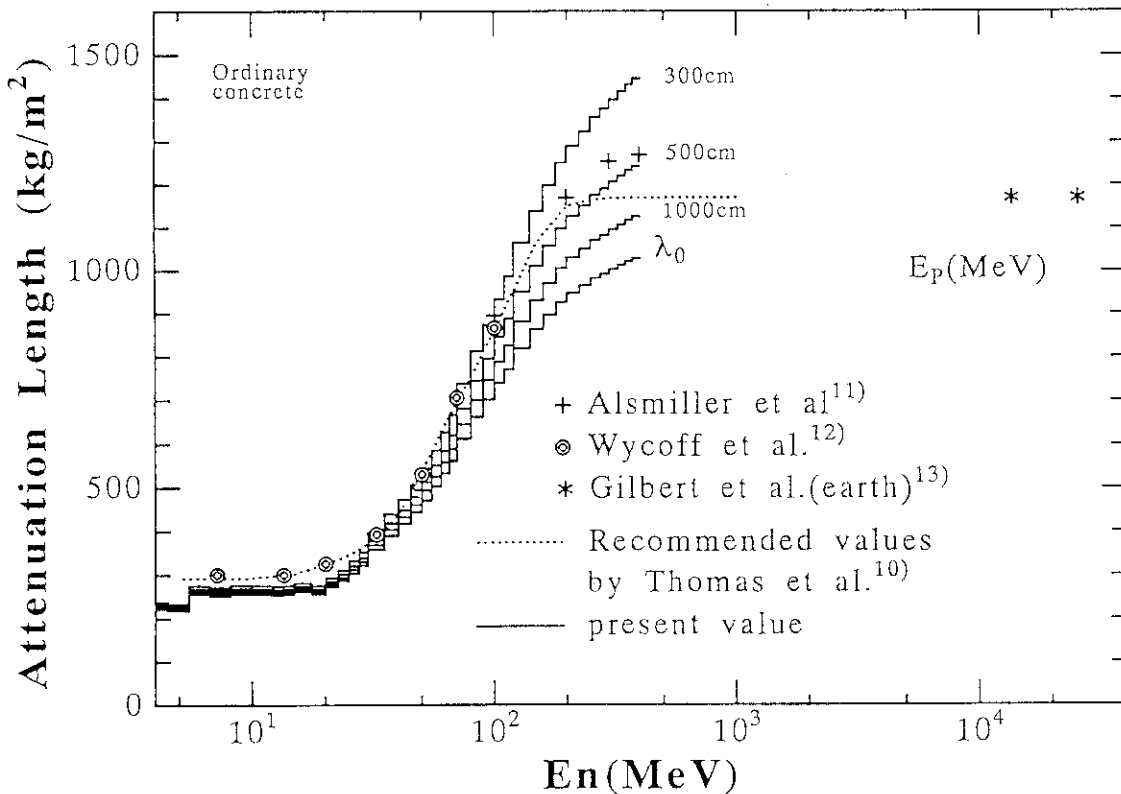


Fig.4.2 Energy dependence of attenuation length obtained at different thicknesses in ordinary concrete, in which other data are compared. Gilbert et al.'s data denotes for 14.6 and 26.4 GeV/c protons.

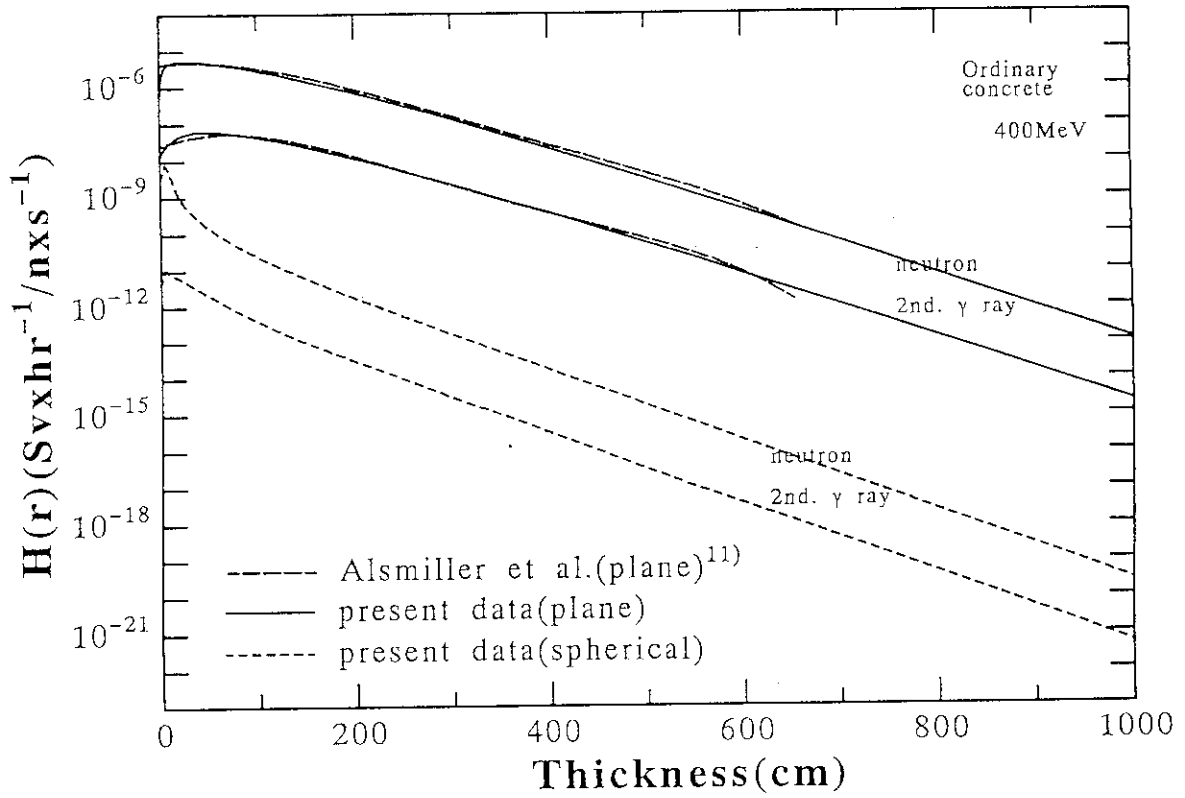


Fig.4.3 Comparison with Alsmiller et al.'s data for dose equivalent in plane geometry.

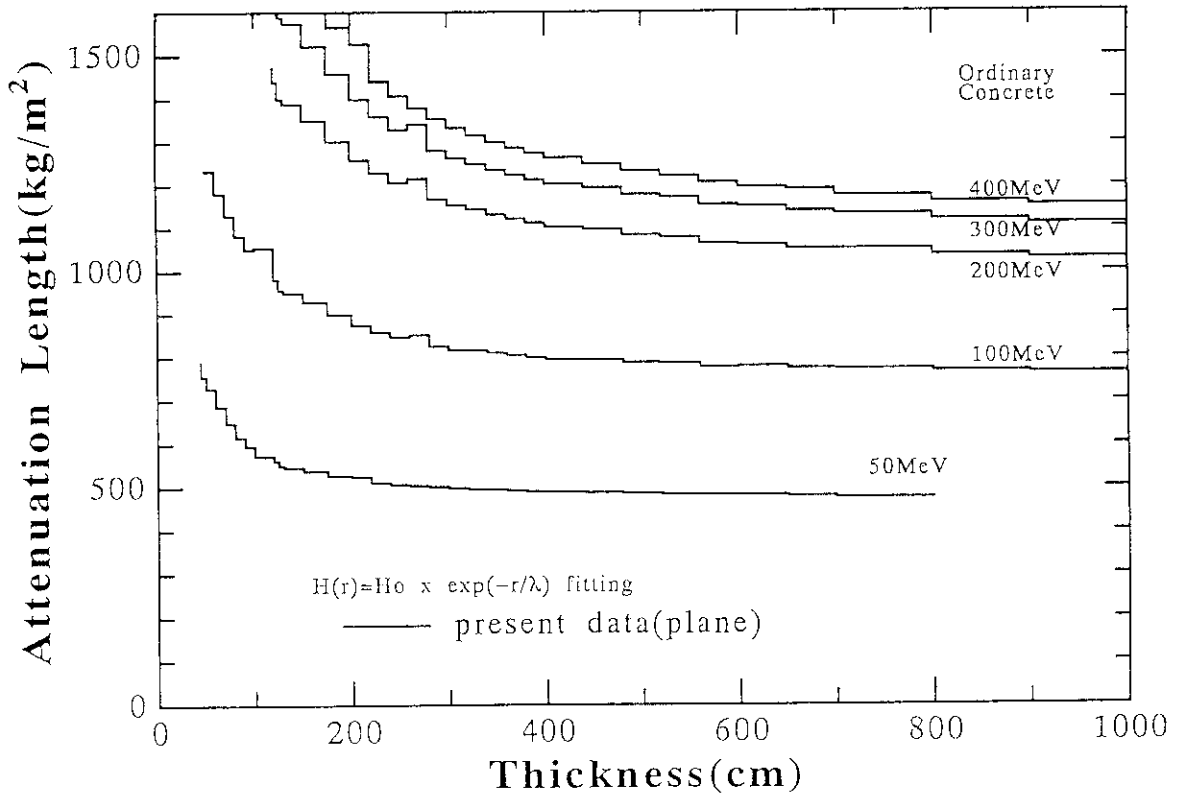


Fig.4.4 Attenuation length of ordinary concrete for different neutron energies in plane geometry.

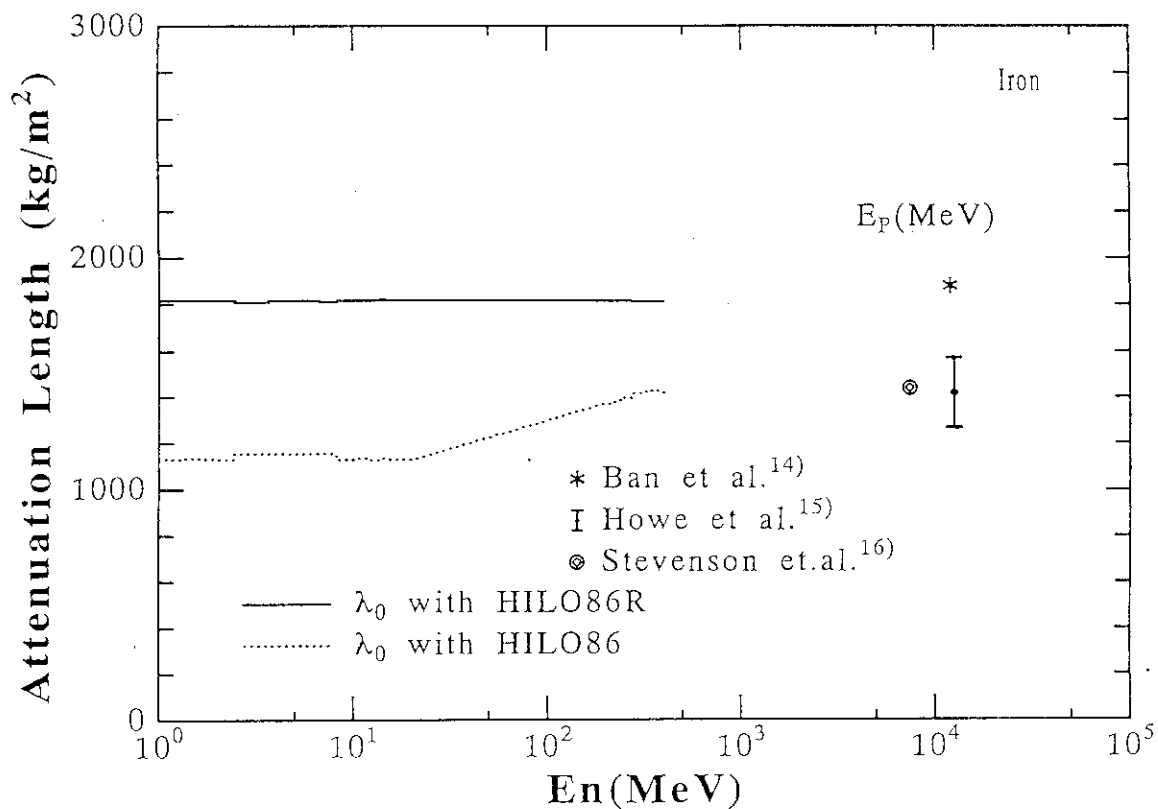


Fig.4.5 Energy dependence of attenuation length obtained in iron, in which other data are compared. Experimental data by Ban et al. and Howe et al. and evaluated data by Stevenson et al. were given for incident proton energy in MeV.

## Appendix

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Table A.1 Attenuation lengths in concrete (from refs. [0] and [11])

	Incident particle		Shields		De tec tor	$\lambda_{att}$ [gcm <sup>-2</sup> ]	Ref.	Remarks
		Beam energy [GeV]	material	density [gcm <sup>-3</sup> ]				
BNL	p	1.5	concrete	4.~4.3	CT	130±15	[1]	
BNL	p	2.5	concrete	4.~4.3	CT	169±32	[1]	
CERN	p	24	concrete	2.5	G5	145±10	[2]	wide beam
CERN	p	20	heavy con	3.6	G5	115±15	[2]	narrow beam
UCRL	p	6.2	concrete	2.4	<sup>11</sup> C	108±20	[3]	
UCRL	p	6.2	concrete	2.4	<sup>27</sup> Al	112±20	[3]	
UCRL	p	6.2	concrete	2.4	<sup>198</sup> Au	116±20	[3]	
ANL	p	12.5	concrete	3.8	IC	173±18	[5]	
CERN	p	15 27	earth	2.2	<sup>11</sup> C <sup>27</sup> Al	117±2 117±2	[6]	
KEK	p	0.74	concrete	3.5	MP	120	[9]	
KEK	p	12	concrete	2.35	MP	143	[11]	

CT: Counter Telescope,/<sup>11</sup>C, <sup>27</sup>Al, <sup>32</sup>S.: Activation detectors,  
 /G5: Nuclear emulsion, /IC: Ionisation Chamber,  
 /MP: Estimated value for Moyer Parameter

Table A.2 Attenuation lengths in iron (from refs. [0] and [11])

	Incident particle		Shields		De tec tor	$\lambda_{att}$ [gcm <sup>-2</sup> ]	Ref.	Remarks
		Beam energy [GeV]	material	density [gcm <sup>-3</sup> ]				
CERN	p	20	steel	7.8	G5	120±10	[4-II]	beam axis
CERN	p	20	steel	7.8	G5	185±20	[4-II]	Integ.beam
CERN	p	10	steel	7.8	G5	119±5	[4-III]	beam axis
CERN	p	10	steel	7.8	G5	165±	[4-III]	Integ.beam
CERN	p	10 & 20	steel	7.8	IC	155±16	[4-IV]	
CERN	p	10 & 20	steel	7.8	MC	128±8	[4-VI]	beam axis
CERN	p	10 & 20	steel	7.8	MC	185±10	[4-VI]	Integ.beam
ANL	p	12.5	steel	7.8	IC	142±15	[5]	
KEK	p	12	steel	7.0~7.2	<sup>11</sup> C	188±12	[10]	
CERN	p	7	steel	7.8	MP	147±10	[7]	
RHEL	p	7.4	steel	7.8	MP	165	[8]	
KEK	p	0.74	steel	7.5	MP	160	[9]	

G5: Nuclear emulsion,/MC: Monte Carlo calculation,  
/<sup>11</sup>C,<sup>27</sup>Al,<sup>32</sup>S,: Activation detectors,/IC: Ionisation Chamber,  
/MP: Estimated value for Moyer Parameter

## References

- [0] R.H. Thomas:"Proton Synchrotron Accelerators",Engineering Compendium on Radiation Shielding,I,56(1968).
- [1] S.J. Lindenbaum,Ann.Rev.Nuc.Sci.11,213(1961).
- [2] A. Citron et al.,Nucl.Inst. and Methods 14,97(1961).
- [3] A.R. Smith:UCRL-11331(1964).
- [4] H. Bindewald et al.,Nucl.Inst.Meth. 32,45(1965).
- [4-I] J. Geibel et al.:"Part I:General Description of the Experiment", N.I.M.32,pp.45-47(1965).
- [4-II] A. Citron et al.:"Part II:A Study of the Nuclear Cascade in Steel by 19.2 GeV Protons",N.I.M.32,pp.48-52(1965).
- [4-III] R.L. Childers et al.:"Part III:Measurement of the Nuclear Cascade at 10 GeV/c with Emulsion",N.I.M.32,pp.53-56(1965).
- [4-IV] J. Baarli et al.:"Part IV:Measurements of A 10 and 19.2 GeV Proton Beam in Steel Using Ionization Chambers and Plastic Phosphors",N.I.M.32,pp.57-60(1965).
- [4-V] L. Hoffmann et al.:"Part V:Studies of the Shielding Required for the Secondary Radiation Produced by a Target in a High-Energy Proton Beam",N.I.M.32,pp.61-64(1965).
- [4-VI] J.A. Geibel and J. Ranft:"Part VI:Monte Carlo Calculation of the Nucleon Meson Cascade in Shielding Materials",N.I.M.32, pp.65-69(1965).
- [5] H.J. Howe Jr., et al.:"On the Shielding of the External Proton Tunnel Area of Argonne's Zero Gradient Synchrotron",ANL-7273(1966).
- [6] W.S. Gilbert et al.:"1966 CERN-LBL-RHEL Shielding Experiment at the CERN Proton Synchrotron",UCRL-17941(1968).
- [7] G.R. Stevenson et al.,Health Phys. 43 13(1969).
- [8] K.B. Shaw and G.R. Stevenson:"Radiation Studies Around Extracted Proton Beams at NIMROD",IEEE.Trans.Nucl.Sci. NS-16 570(1969).
- [9] R.H. Thomas:"Transverse Shielding for KENS:II",Studies on Current Problems of Radiation Shielding in KEK,KEK-78-7,pp.15-28(1978).
- [10] S. Ban et al.,Nucl.Inst.Meth.174,271(1980).
- [11] S. Ban:"Studies on the Shielding of High Energy Proton Synchrotron",Report at KEK(1982),in Japanese.