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Progress Report of Shielding Investigations in Japan

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Progress Report of Shielding Investigations in Japan

Summary

This report contains the main results of shielding studies made in Japan in two years since the spring of 1964, and is a continuation of our previous report JAERI-4029 (1964). Theoretical and experimental researches of gamma-ray problems, neutron problems and duct streaming problems as well as the state of the mockup tests of Japanese nuclear ship are described.

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日本における遮蔽研究の成果報告

要　　旨

１９６４年春以降の２ケ年間における，日本での遮蔽研究の成果についてまとめてある。ガンマ線問題，中性子問題，ダクトストリーミング問題について理論的，実験的研究をのべてあるほか，原子力第１船模型実験の状況も報告してある。本書は JAERI－４０２９（１９６４）の続刊である。

１９６６年７月

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PREFACE

In March 1964, the first international panel meeting of the reactor shielding was held in Vienna sponsored by IAEA, and valuable informations were presented from many countries of the world. The "Status Report of Shielding Investigation in Japan" (JAERI-4029(1964)), which was presented at this meeting, describes the results of researches obtained in the field of shielding till the beginning of 1964, as well as the brief outline of the JRR-4, the swimming pool type reactor installed in JAERI especially for the shielding studies.

Two years have passed since the meeting and some new results of shielding researches were obtained. JRR-4 was begun to be used for the experimental works. It may be significant at this time to collect these results in the past two years and to publish in the form of a report. This report is,therefore, a continuation of our previous report JAERI-4029.

In the 1966 Annual Meeting of the Atomic Energy Society of Japan, a special lecture was given by Y. Tanaka on the progress and future course of our investigation. Prior to this lecture, earnest discussions were done by several members of the shielding researchers group. They were Yoshihisa Tanaka of Kawasaki Dockyards, Mitsuyuki Kitazume of Hitachi Ltd., Takayoshi Fuse of the Ship Research Institute, Iwao Umeda of Nippon Kokan Ltd., Akira Tsuruo and Mitsuo Shindo of JAERI. This report is written on the discussions by above listed members.

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CONTENTS

I. INTRODUCTION

A summary of the shielding investigation in Japan till 1964 is reported in JAERI-4029, and considerable differences are found in many respects when the state of researches described in the report is compared with that of two years after 1964. The main features are (1) more precise and exhaustive studies of gamma-ray problems were done than before, (2) researches on the neutron transmission problems began, (3) duct streaming studies were promoted theoretically and experimentary, and (4) mockup tests for the nuclear ship of Japan started.

Before 1964, in the field of experiment, the neutron sources for shielding study were rather difficult to obtain, and great efforts were concentrated on the gamma-ray problems, because the radioisotopes as gamma-ray source were readily available. Since reactors became possible as radiation source for shielding studies, the neutron transmission through shield barriers came to be the matter of concern for many researchers. In the field of theoretical studies, the development of computer code continued, and the comparison of calculated results with the experimental data obtained in Japan began. Irregularities in the shields cause a trouble in any type of reactor plant, and no reliable method of calculation nor sufficient data for practical design are at hand. A systematic study for rather fundamental geometry is continued in this field. Since the JRR-4 came in operation, the large scale samples of shield materials became possible to be tested, and, in conjunction with the development of nuclear ship of Japan, mockup tests started.

The following sections contain the main results of researches in each field of shielding since 1964.

II. GAMMA-RAY PROBLEMS

The important problems in the behavior of gamma-rays are the transmission through multilayers and the albedo, and our main effort is concentrated on these two problems. For the multilayer transmission, approximate calculation, accurate solution and empirical formula for buildup factors were studied, and the comparison with experiment confirmed that reliable method of calculation had been developed. In the albedo problem, differential albedo was studied both theoretically and experimentally and fruitful results were obtained.

2.1 Theoretical Studies

2.1.1 Monte Carlo Method (BREVI[1])

Broder et al.[2] obtained a semi-empirical formula for the dose buildup factors of gamma-rays in the case of point isotropic source. This formula, however, dose not agree with our understandings, and some theoretical and experimental tests are considered necessary. The BREVI code was developed to treat the gamma-ray transmission problems through double layers including the geometry of Broder's experiments. The source conditions of this code are point source and plane source, in each of which isotropic, mono-directional and cosine distributions are possible to set. To get the results of calculation for thick shield barriers rapidly, the systematic sampling technique for the first collision of gamma-ray is applicable as well as the usual sampling technique (for cosine distributed source, this choise is not allowed). As mentioned in 2.2.1, the calculated results for lead-graphite and graphite-lead layers agree with the experimental results[3] up to 7 - 9 mean free paths. The results for single layer also agree with the experiment.[4] In the case of plane isotropic source, the comparison with EOS method[5], mentioned below, is also possible.

2.1.2 Response Matrix Method[6]

Originally, this method is introduced as an improvement on the Monte Carlo method to overcome the inefficiency of the latter. To evaluate the gamma-ray transmission through multilayers, an operator is prepared for an elementary part of shielding barrier by the Monte Carlo method. This operator should include all information to synthesize the result for the whole shield. In the case of slab geometry, the energy and angular direction of a photon are needed to present the photon condition, and, in general, the operator has a form of matrix. The response matrix method is, accordingly, a combination of operator method and Monte Carlo method. The computation time to perform a synthesis calculation for a layer of shield slab is short, and the response matrix method is powerful for the practical design of gamma-ray shielding. The response matrices for water, aluminum, iron and lead have been prepared by the technique employed by Berger[7], and not only the total energy flux or current but also energy angular and/or angular spectrum can be obtained by this method. The penetration of gamma-rays through five layers of lead, iron and water was treated by this method. Some results of calculation are shown in Figs. 1 and 2. In Fig. 2 the calculated results by the response matrix method are compared with the experimental[8]. This code is prepared for the NEAC-2206 computer.

2.1.3 Legendre Polynomial Method (EOS-1[4] and SELENE[9])

To solve the Boltzmann transport equation as exactly as possible in the case of slab geometry, the computational method retaining the advantage of NIOBE[10] was developed. The angular flux is expanded in a series of Legendre polynomials and the iteration process is utilized for the spatial integration of the equation, as in NIOBE. For the case of gamma-ray transmission problem, the calculated results by this method of evaluating slab shield, EOS-1, were compared with those by the moments method(11) and Monte Carlo method(12). A good agreement was found both in differential energy spectrum and in energy buildup factor as shown in Figs. 3 and 4. Figure 5 shows the angular distribution and energy spectrum compared with the experiments(13).

Although an accurate solution is obtained by the EOS method in the case of problems which have no sharp peak in the angular flux, some troubles are found in the treatment of problems which have sharp peaks in the flux as in the case of mono-energetic, mono-directional source. To avoid this difficulty, SELENE code was developed. In this code, the Sn method is applied in the numerical integration of Boltzmann equation. Calculated and experimental results are compared in Fig. 6.

EOS-1 and SELENE codes are prepared for the NEAC-2206 computer.

2.1.4 Invariant Imbedding Method

The invariant imbedding principle, originally introduced in the field of astrophysics, is applied to study the transmission and reflection of gamma-rays through homogeneous slab shields of finite thickness(14). Numerical solutions are obtained for the albedo, and good agreement with Monte Carlo solutions(15) is found as shown in Figs. 7 and 8. A fair agreement is also obtained with the experimental results(16) for the energy spectra of reflected gamma-rays. For the angular distribution of reflected gammas from light elements, however, the agreement is rather poor. See Figs. 9 and 10. Deep penetration is also being studied(17). In this method, the fundamental equation is non-linear type in general, and some mathematical techniques are needed to get a solution. Multilayer problem is not yet treated by this method.

2.1.5 Miscellaneous Studies

A simple calculation method is applied to the gamma-ray transmission through shield barrier[18], in which the photons normally incident on a shield slab are assumed to suffer single scatterings in the shield and emerge from the rear face of the slab. The total number and energy flux transmitted are obtained by integrating over the emergent angle. Energy buildup factors for lead, iron, concrete and water were calculated from 1 to 4 mean free paths, and calculations were also made for double layers of lead-water shields. The calculated results were compared with those of Monte Carlo method and moments method, and good agreement was found, considering the simplicity of the method, especially for lead. As the multiple scattering effect is ignored, the buildup factors in this method are smaller than that of Monte Carlo method.

For the case of point isotropic source, the number and energy build-up factors and energy current spectra of gamma-rays transmitted through spherical and cylindrical shields, and angular distribution through spherical shields were calculated by Monte Carlo method[19]. A spectrum of absorbed radiation, emitted from a ^{60}Co point isotropic source placed at the center of a cylindrical plastic scintillator was measured by a scintillation spectrometer for comparison with the results obtained by this method, and good agreement was found as shown in Fig. 11.

The buildup factor for the gamma-rays penetrating the multilayers is very important in the practical design. Semi-empirical formula of Broder et al.[2] do not show good agreement with the experimental results[3,20], especially in the case of the double layers of heavy material after light material. A modification of Kalos's formula[21] was introduced[22], and fair agreement with the experimental buildup factors were obtained in the gamma-ray energy range from 1 to 8 MeV for double layers.

The Rockwell's method for calculating the unscattered radiation flux from sources of spherical or cylindrical shape has the disadvantages of onerous procedure, inaccuracies of approximation and limitations on the available range. An approximate method based on the modyfication of the source geometry to a form convenient for calculation is proposed(23). The sphere is approximated by a conical fragment of spherical shell, and the cylinder by a columnar fragment of cylindrical shell. Graphs and accuracies of the functions for approximate calculation were obtained. The effect of shell-shaped shield surrounding a spherical or cylindrical source is also obtained(24). The exact equations and their approximate calculation formulas were derived. The graphs of the correction function and the accuracies of this method were obtained and the comparison with experimental results were shown.

2.2 Experimental Studies

2.2.1 Multiple Layers

Using the same method as Kirn et al., the buildup factors of multilayers of water, iron and lead for ^{60}Co monodirectional source were measured(20). An empirical formula was proposed, and it was found that by this formula the buildup factors could be obtained for multilayers composed of a combination of materials by using known buildup factors of the monolayers of composing materials. Also by Kirn's method, for ^{60}Co monodirectional source the buildup factors for double and triple layers of lead, iron, aluminum and concrete were measured(8) and Kalos's formula was found to be applicable in the cases of double layers of lead and lighter materials, especially when lead was backed by the lighter material. These experiments covers the gamma-ray mean free paths as far as six.

Energy spectrum of gamma-rays transmitted through multilayers of iron and polyethylene were measured by 5" x 5" NaI scintillator for the plane monodirectional source of ^{60}Co[25]. Pulse height was converted to energy spectra by 19 x 19 inverse matrix. The experimental results agreed well with the Raso's calculated ones.

As pointed out in 2.1.1, Broder's semi-empirical formula for the dose buildup factors of gamma-rays from the point isotropic source is considered to have some problems to be tested, so measurements were done using the pulse dosimeter[26] for just the same geometrical condition as Broder et al's. The Monte Carlo calculation was performed using the above mentioned BREVI code. As shown in Fig. 12, Broder's formula seems to do not coincide with the experimental results.

Using an uranium converter plate installed in TRIGA reactor of St. Paul's University as source, the gamma-ray attenuation by water-iron double layer shields was measured by ion chamber[27]. The source intensity was not so strong that no experiments of the deep penetration could be made.

2.2.2 Backscatterings

The scattering phenomenon of gamma-ray at the boundary of shield barrier is another important problem as the multilayer buildup factors. A series of measurements of backscatterings were performed and the data of number and energy albedo obtained. Data of differential albedo were also obtained.

As mentioned in the previous section, experimental results[17] of the energy albedo for the ^{60}Co and ^{137}Cs point source were compared with the calculation by invariant imbedding method and good agreement was found. The correlation of the thickness of scatterer and albedo, using the slab of polyethylene and aluminum as the scattering shield slab, was measured[28], and by the data of backscattered gamma-rays in the lead slab, it was confirmed that these gamma-rays were composed mainly of singly scattered ones(29). To find the regions where the gamma-rays were backscattered the most, the distribution of gamma-rays backscattered normally from the slab of polyethylene, iron, aluminum and lead were measured and the analysis of the data is now in progress(30). The backscatterings in the NaI(Tl) scintillator crystal were also investigated(31).

2.2.3 Miscellaneous Studies

In the experimental studies of buildup factors, point isotropic sources are often utilized and Kirn's method is also used for plane monodirectional source geometry. For line isotropic source geometry, an alternative arrangement of a point isotropic source and long cylindrical ionization chamber was employed(32). The dose buildup factors for transmitted gamma-rays from infinite line isotropic source through an infinitely extensive slab shield placed in parallel with the source were obtained.

For the purpose of obtaining the dose buildup factors from the volume source, 1 mm x 1 mm ^{60}Co pellets (about 50μc) contained in 3 cm diameter polyester balls, were packed in a plastic cylinder(33). Linear absorption coefficient of the source material was found to be 0.076 cm. Cylindrical shell of iron and water was used as the shield barrier and dose buildup factors measured by pulse dosimeter(26) are shown in Fig. 13.

Structural components of ship have shielding effects for nuclear radiations. In the calculation of these effects, the buildup factors of gamma-rays computed for each component member are superposed so as to get the total buildup factors of the structural components. In order to compare these computed results with the actually measured data, the gamma-ray dose measurements were done using the ^{60}Co sources and gamma-ray surveymeters(34). Shintoku-maru, a training ship of about 3,000 gross tons, was selected for this purpose. For about 200 points of measurement, the difference between the computed and measured values became larger with the increase distance between source and the measuring point.

III. NEUTRON PROBLEMS

In the design of reactor shields, the neutron transmission through multilayers is the most fundamental problem. The neutron transport phenomenon is far more complicated than gamma-ray transport, and it is very difficult to treat it accurately for neutron energies up to thermal values. Though it is not so long since the investigation of neutron problems was started in Japan, reactor sources and medium and large scale computers became available and some studies have been done in both theoretical and experimental fields.

3.1 Theoretical Studies

3.1.1 Removal-diffusion Method (RAC)

RAC code was originally prepared for IBM-7090, and then translated to 7044. RAC can handle the cylindrical source geometry with cylindrical shell shields. 7044 RAC are composed of the following sub-codes: REMOVAL computes energy-dependent removal group neutron distributions up to 18 energy groups. DIFFUSION calculates up to 6 energy groups. PRY.GAMMA can evaluate the primary gamma-ray distribution in energy groups up to 10 including the buildup factor of each group, and in SEC.GAMMA the source distribution is determined by the thermal neutron flux of DIFFUSION. REMOVAL, DIFFUSION and SEC. GAMMA can be used in one run as well as separately. The RAC code is used extensively and the results of comparison with experimental data are described in the following section.

3.1.2 Legendre Polynomial Method (EOS-2[35])

EOS-2 was prepared for neutron transmission problems and computational method and restrictions are same as EOS-1. Fig. 14 shows the neutron differential number flux for fission source calculated by EOS-2 method and moments method, and differential angular number spectrum together with the experimental results of BSR-1.

3.1.3 Photoneutron Analysis[36]

In a water-shielded reactor, photoneutrons are distributed in the water layer in addition to the neutrons produced in the core, and the observed neutron flux distribution has a bend caused by the contribution of photoneutrons. Simple analysis using two group diffusion theory is developed, and it is shown that this method can reproduce the photoneutron distribution reasonably well in a region where the neutron flux originating from the reactor core is negligibly small, and that the distribution of photoneutron flux together with that of the core neutron flux explain the observed bend of BSR-1.

3.2 Experimental Studies

3.2.1 Fission Plate Experiment

Fission plate in the lid tank of TRIGA reactor of St. Paul's Univ. was used as a neutron source and the penetration of fast neutron through iron-water multilayers was measured by In, S and Al[37]. The fact that the removal cross section should be considered as a function of neutron energy was assertained. Thermal neutrons were detected before and behind the iron barrier by BF_3 counter and gold foils. The source strength is so small that the deep penetration problem cannot be studied.

3.2.2 HTR Experiments

In the pool of HTR (Hitachi Training Reactor), a series of neutron penetration experiments were done[38,39,40,41,42,43,44]. For various geometries of iron-water multilayers, neutrons were measured by BF_3 counter and activation method (Al, Fe, S, P, Au, In, Cu). The attenuation of neutron flux in the iron barrier was also measured. These experimental results were compared with the calculation using the previously mentioned RAC code[40,41]. See Fig. 15. Good agreement was found as far as 1.5 m of penetration distance, but some disagreements appeared in the deep penetration. The reason of discrepancy is now searched. The removal cross section of iron was also measured, but in this case, the core radiation was used as a neutron source instead of the usual fission plate, the correction was introduced[39]. The removal cross sections of ordinary and heavy concretes were measured, too[45].

3.2.3 Accelerator Experiment

Fast neutrons from d-D (2.5 Mev) and d-T (14 Mev) reactions were introduced into a water tank containing the shield slab of iron, aluminum and concrete, and the attenuation of thermal neutron was measured by foils and BF_3 counters[41,46]. The removal cross section of the shield was found to be energy dependent.

3.2.4 JRR-4 Experiment

In the design of marine reactor system, the weights and spatial arrangements of the shields are very important as well as the economy and the radiation damage of materials used, and often the weight of shields play a decisive role. The Ship Research Institute group is now carrying a series of neutron penetration experiments in the No.1 pool of JRR-4. 1.5 m x 1.5 m iron slabs of thickness 3, 5, 10 and 15 cm constitute an iron-water multilayer, and neutrons were detected by the threshold detectors (Mg, Al, Fe, Zn, P, and In) and resonance detectors (Al, V, W, Mn, Cu, Au and In). Optimization study is planned. Photo. 1 shows the experimental iron slabs in the No. 1 pool of JRR-4.

IV. DUCT STREAMING PROBLEMS

4.1 Theoretical Studies

Systematic studies of the gamma-ray leakage through cylindrical duct were made(47,48). Infinite plane source on the side of shield barrier is separated in two parts, a disk source at the entrance of the duct and an annular source surrounding the entrance of the duct. The unscattered and singly scattered flux from the former is obtained by integration, and the multiple scattering flux is calculated by the Monte Carlo method utilizing partly the results of the integration. The ray-analysis method including the buildup factor is used to get the flux from the latter. The main conclusions from these studies are as follows:

1. Most of the gamma-rays which started the disk source, reflected at the inner surface of duct and reached the exit, are singly scattered radiation from the shallow part of the duct wall near its surface, and its contribution to the total dose at the duct exit is about 40% at most, and becomes smaller as the duct length becomes larger.

2. The contribution of the annular source is not negligible, when the source region extends widely around the duct entrance, and is predominant when the duct length is small. The effective radius of the source region contributing to the exit dose is calculated.

For isotropic and cosine angular source, graphs available in the practical design were obtained for some typical shield materials. The details of the calculated results will be published shortly.

Ray analysis method in the gamma-ray problems is considered to be useful also in the fast neutron problems, and the calculation of neutron streaming problem is now under study(49,50). But some techniques should be introduced to get the solution for the thermal and epithermal neutrons streaming through the duct.

4.2 Experimental Studies

In the lid tank of TRIGA reactor of St. Paul's University, the streaming of neutrons and gamma-rays through pipes of polyethylene, vinyl-chloride, iron and lead was measured[51,52]. These ducts were placed normally and/or parallel to the source plate, and in some cases, lid of lead was set at the duct entrance and exit. It was found that the behaviors of fast, epi-thermal and thermal neutrons are different and separate treatments are necessary in the calculations. The fact that the wall material disturb the intensity and distribution of epi-thermal and thermal neutrons, and that the streaming along the duct wall occurs, was also found. A remakable side streaming is seen when the duct is set parallel to the source plate.

Threshold detectors for fast neutrons, and BF_3 counter, In foil and Mn wire for thermal neutrons were used to measure the streaming through acrylate pipe in the HTR pool[53]. The results are shown in Fig. 16.

In order to test the validity of the previously mentioned theoretical results, a series of experiments of gamma-ray streaming is now carried out in JAERI. The shield materials are lead, iron, and concrete and pulse dosimeter[26] is used as a detector. The preliminary results show good agreement of the theory and the experiment.

V. MOCKUP TESTS

Shielding mockup tests for the first nuclear ship of Japan is now going at JRR-4. JAERI, the Ship Research Institute and the Japan Nuclear Ship Development Agency are in corporation for this tests. The purpose of tests is not to obtain fundamental knowledges of reactor shielding in general, but to obtain the data useful for the design of the ship.

The tests consist of two parts; the primary shield mockup tests and the secondary shield mockup tests. The former is carried mainly in the No. 2 pool and the latter in the dry test facility of JRR-4.

As for the primary shield mockup tests, the measurements of neutrons and gammas in the core radial direction for the layers of thermal shield, reactor pressure vessel and iron-water multilayer shields are planned. The experimental results will be compared with computer code calculations. The shielding effect of nozzle of main steam pipe through reactor vessel and the shielding effect of control rod drive mechanism will be tested and some qualitative conclusions are expected.

The secondary shield mockup tests include the measurements of radiation streamings through slit and duct. Lead duct of 35 cm inner diameter was set in the dry test facility and the gamma-ray from experimental beam hole was introduced in this duct. Photo. 2 shows the experimental arrangement of this mockup. The dose and spectrum of gamma-rays were measured at the exit of the duct. The angle between incident beam and duct axis is variable. Preliminary analysis of the experimental results is now in progress. Lead, concrete and polyethylene and their combinations will be used in the slit experiments. The gap of the slit and beam incident angle are variable.

The tests will be continued for about a year, and the results will be presented in future.

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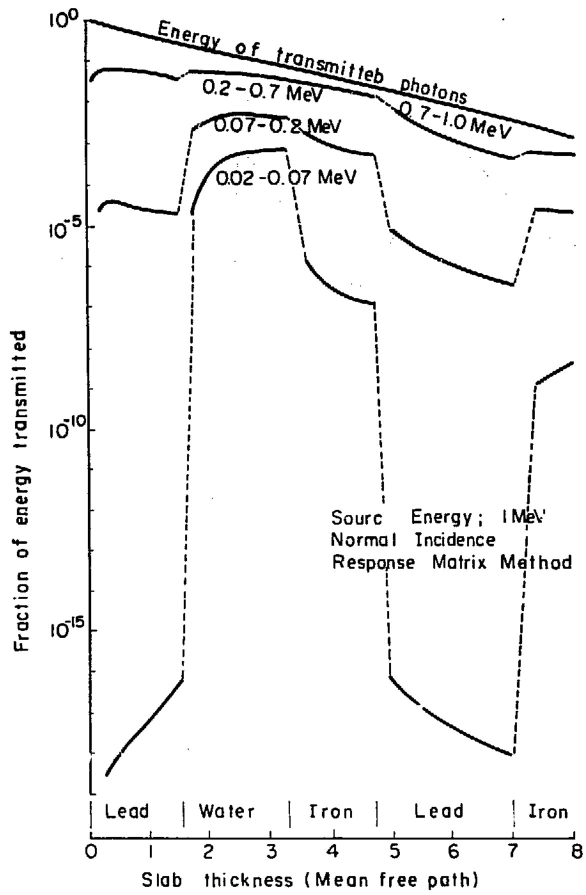
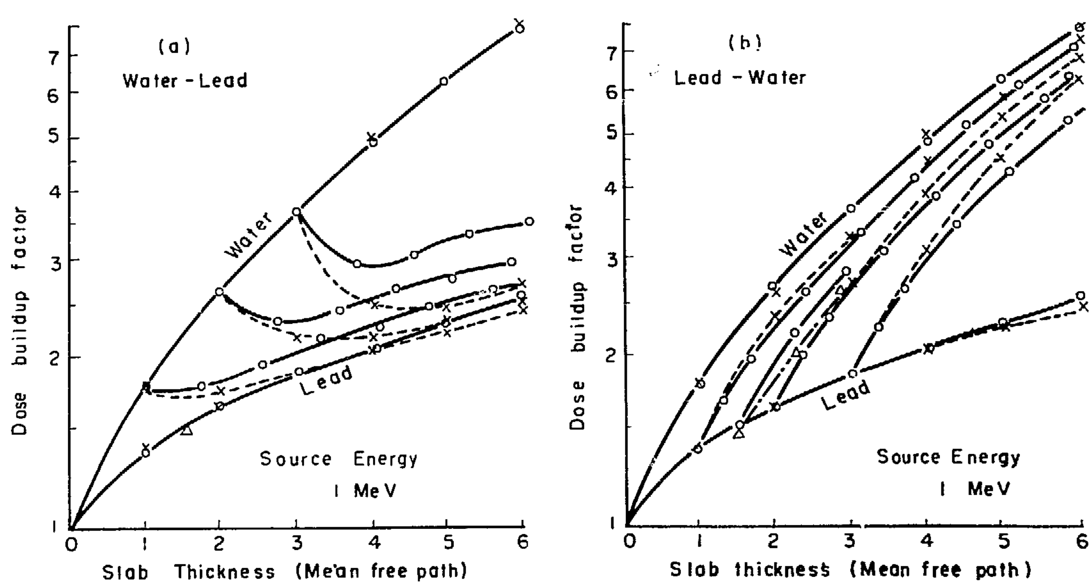


(a) Energy transmission through stratified slabs



—o— RESPONSE MATRIX CALCULATION
--x-- MONTE CARLO CALCULATION
—Δ--- SOLUTIONS BY DIRECT INTEGRATION OF SLAB GEOMETRY TRANSPORT EQUATION COSINE OF INCIDENT ANGLE=0.9863.

(b) Dose build-up factors through stratified slabs (lead and water)

Fig. 1 Computed results by the response matrix method and the comparison with experimental

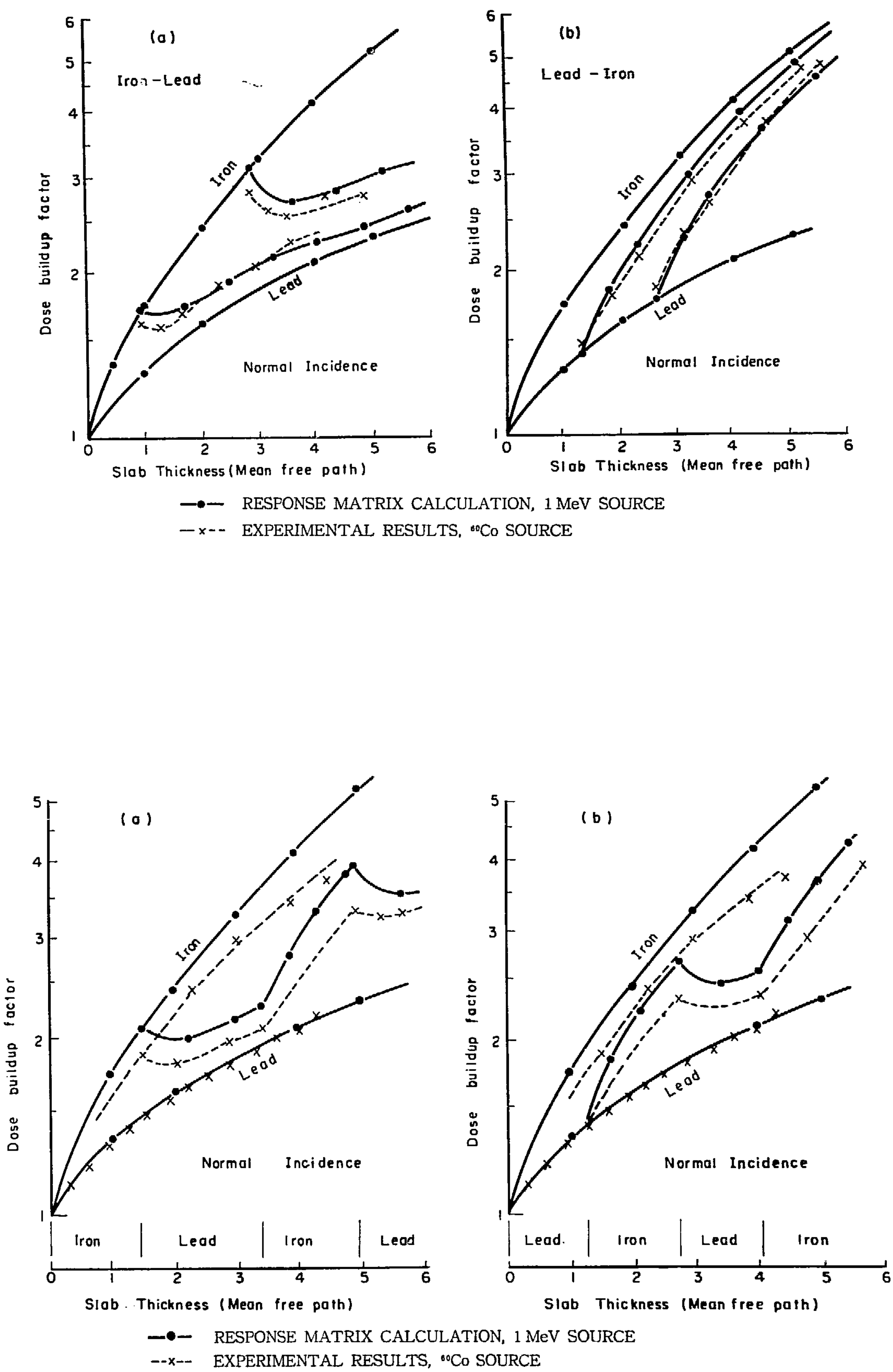
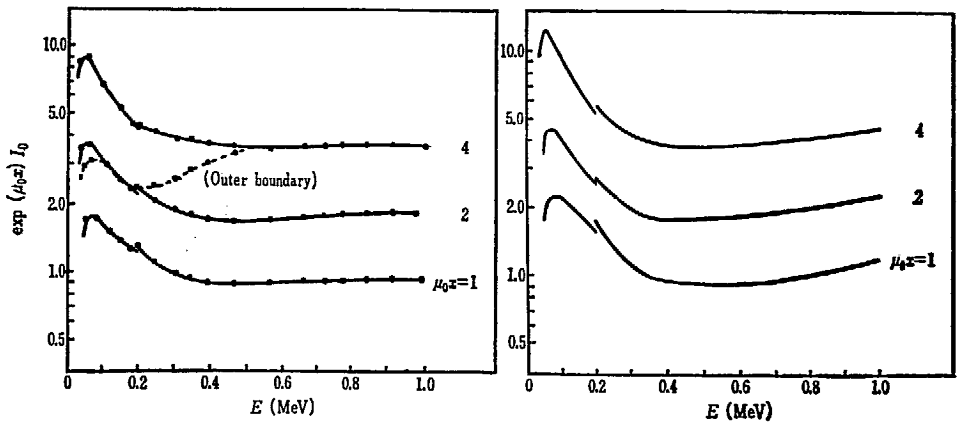
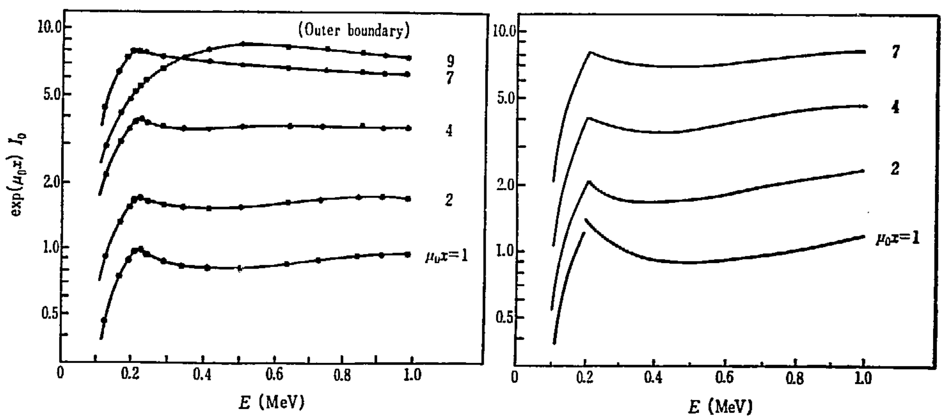


Fig. 2 Dose buildup factors through stratified slabs (lead and iron)



Medium: Water, Source: 1. 0 MeV, plane monodirectional

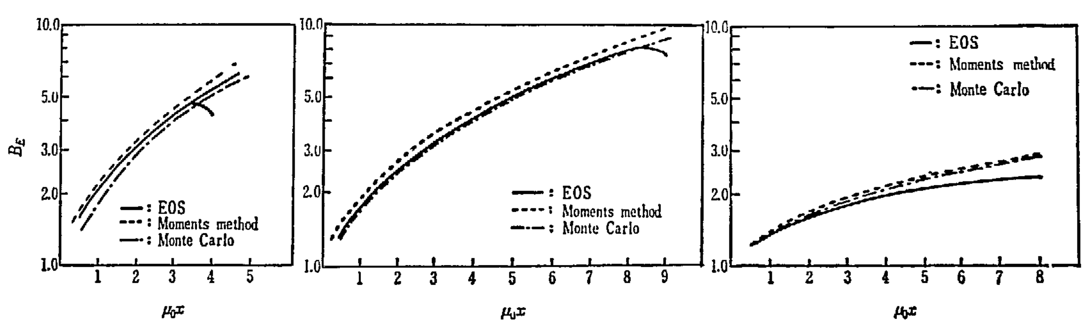
(a) EOS calculation (b) Moments method



Medium: Iron, Source: 1. 0 MeV, plane monodirectional

(a) EOS calculation (b) Moments method

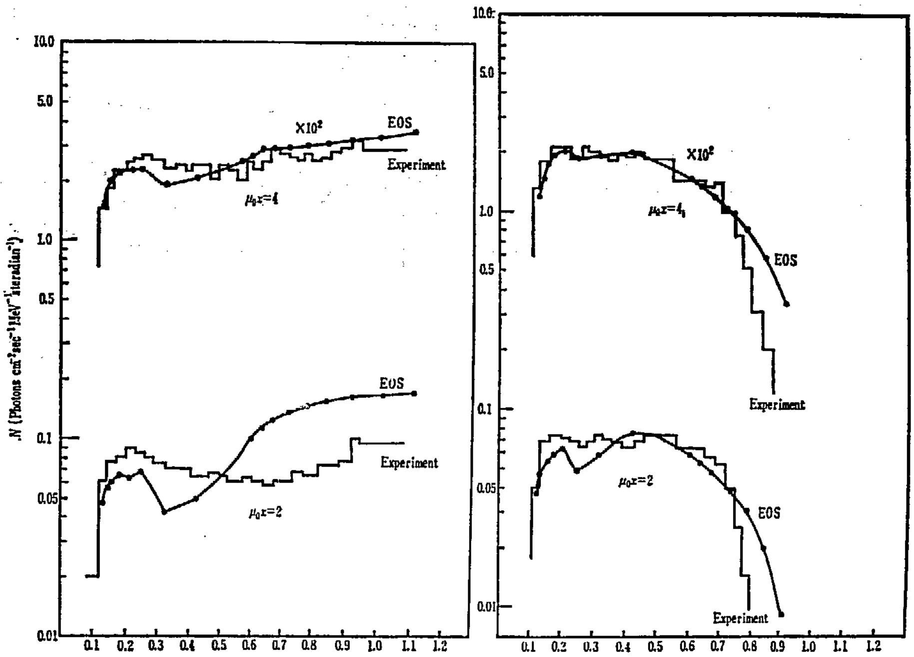
Fig. 3 Differential energy spectrum



Source: 1. 0 MeV, plane monodirectional

(a) Water (b) Iron (c) Lead

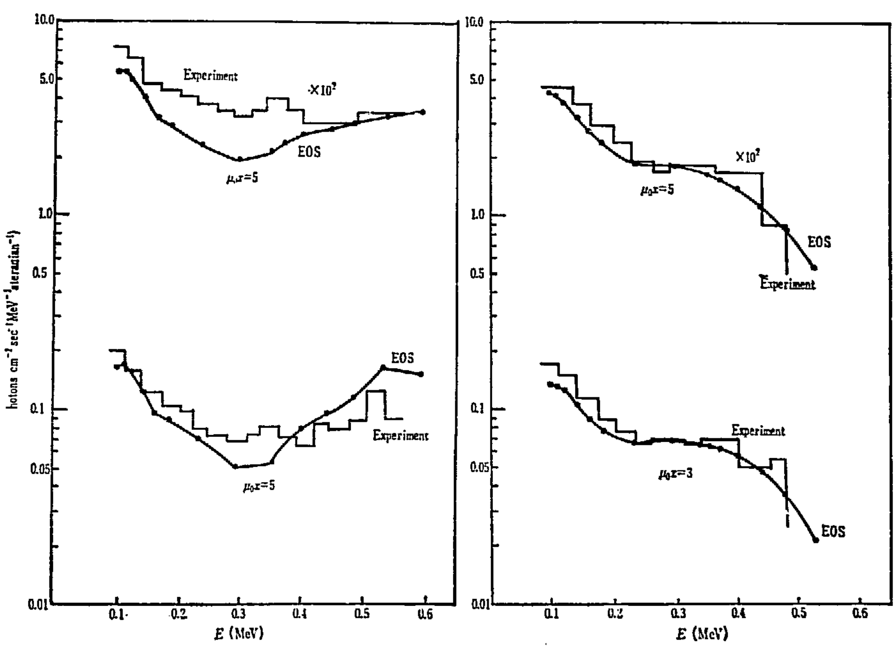
Fig. 4 Energy buildup factor



Material of plate: Iron, Source: ^{60}Co, normal incidence

(a) Transmitted photon angle 20°(Experiment), 24. 7°(EOS)

(b) Transmitted photon angle 60°(Experiment), 61. 9°(EOS)



Material of plate: Aluminum, Source: ^{137}Cs, normal incidence

(c) Transmitted photon angle 20°(Experiment), 24. 7°(EOS)

(d) Transmitted photon angle 60°(Experiment), 61. 9°(EOS)

Fig. 5 Angular distribution and energy spectrum

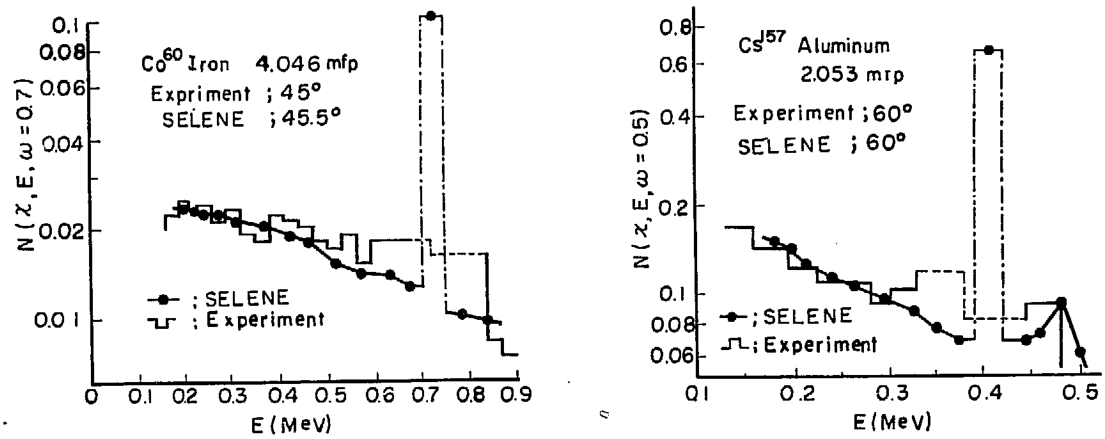
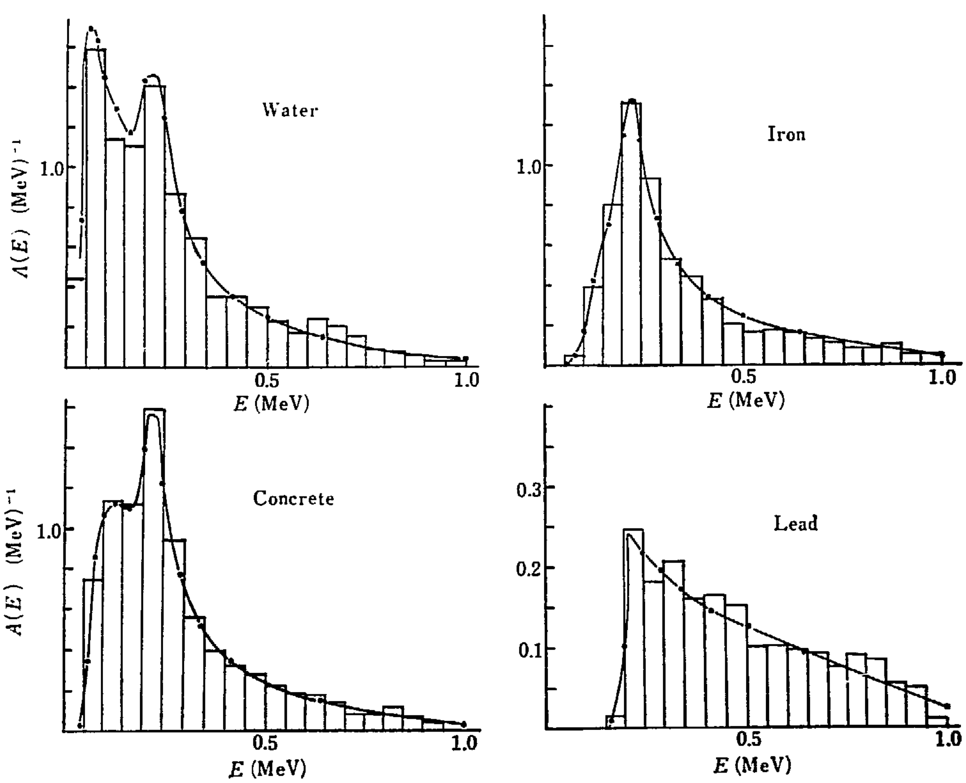


Fig. 6 Compairson of angular spectrum by SELENE method with experiment

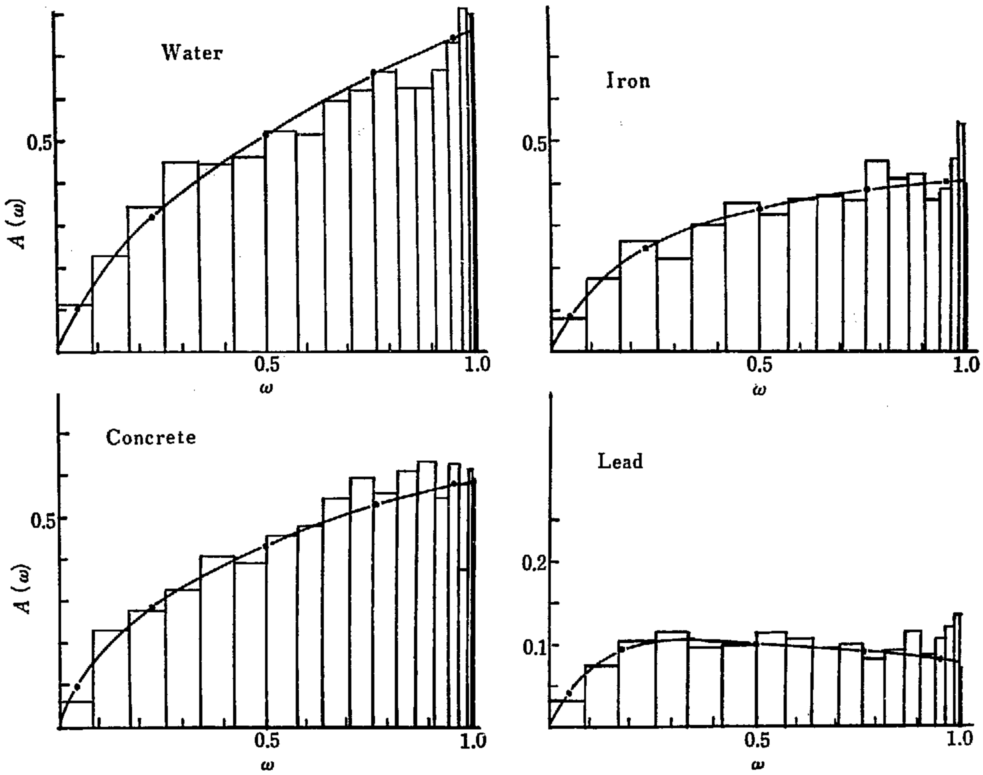


Incident current is isotropic with energy 1 MeV.

⊓ = Monte Carlo calculation by Berger and Raso

• = Present calculation

Fig. 7 Energy spectra of reflected radiations from various substances

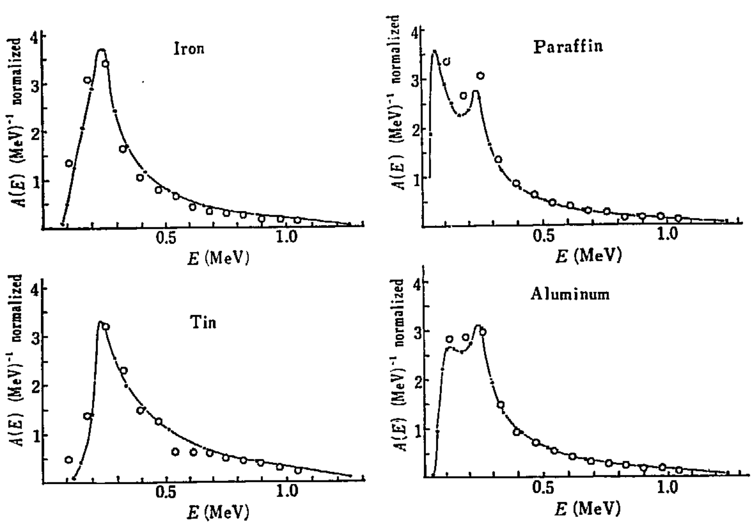


Incident current is isotropic with energy 1 MeV.

⊓=Monte Carlo calculation by Berger and Raso

• =Present calculation

Fig. 8 Angular distributions of reflected radiations from various substances



Source: Isotropic point source of ^{60}Co

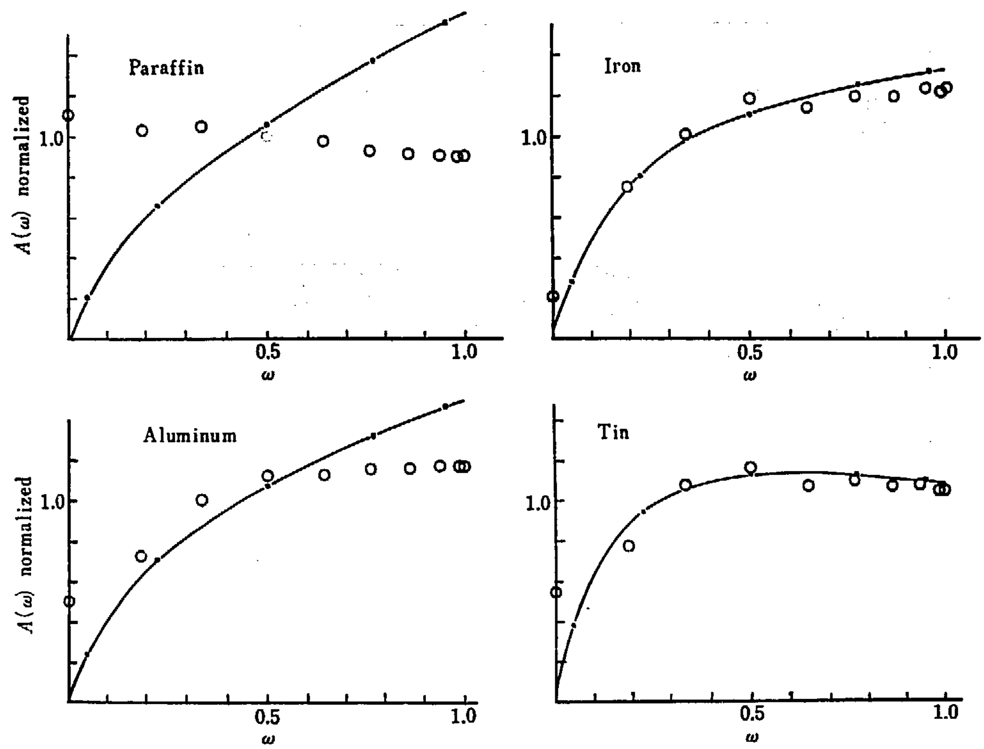
○=Measurement by Hyodo

• =Present calculation

The ordinate is the energy spectrum normalized by the equation.

$$\int_0^{E_0} A(E)dE=1$$

Fig. 9 Energy spectra of reflected radiations



Source: Isotropic point source of ^{60}Co
○=Measurement by Hyodo • =Present calculation
The ordinate is the angular distribution normalized by the equation.

$$\int_0^1 A(\omega)d\omega=1$$

Fig. 10 Angular distribution oi reflected rediations

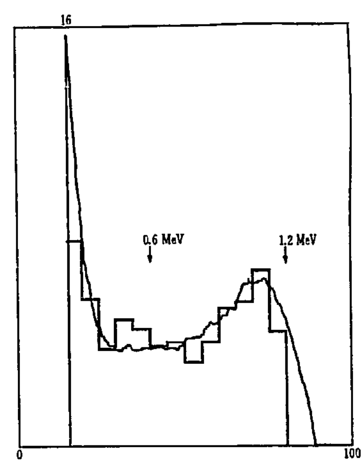


Fig. 11 Comparison of energy spectra in scintillator with Monte Carlo calculation

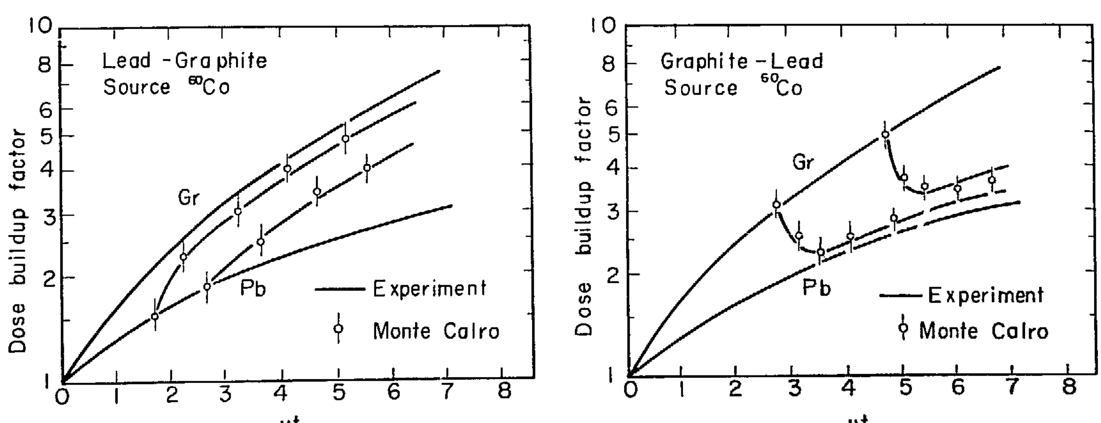
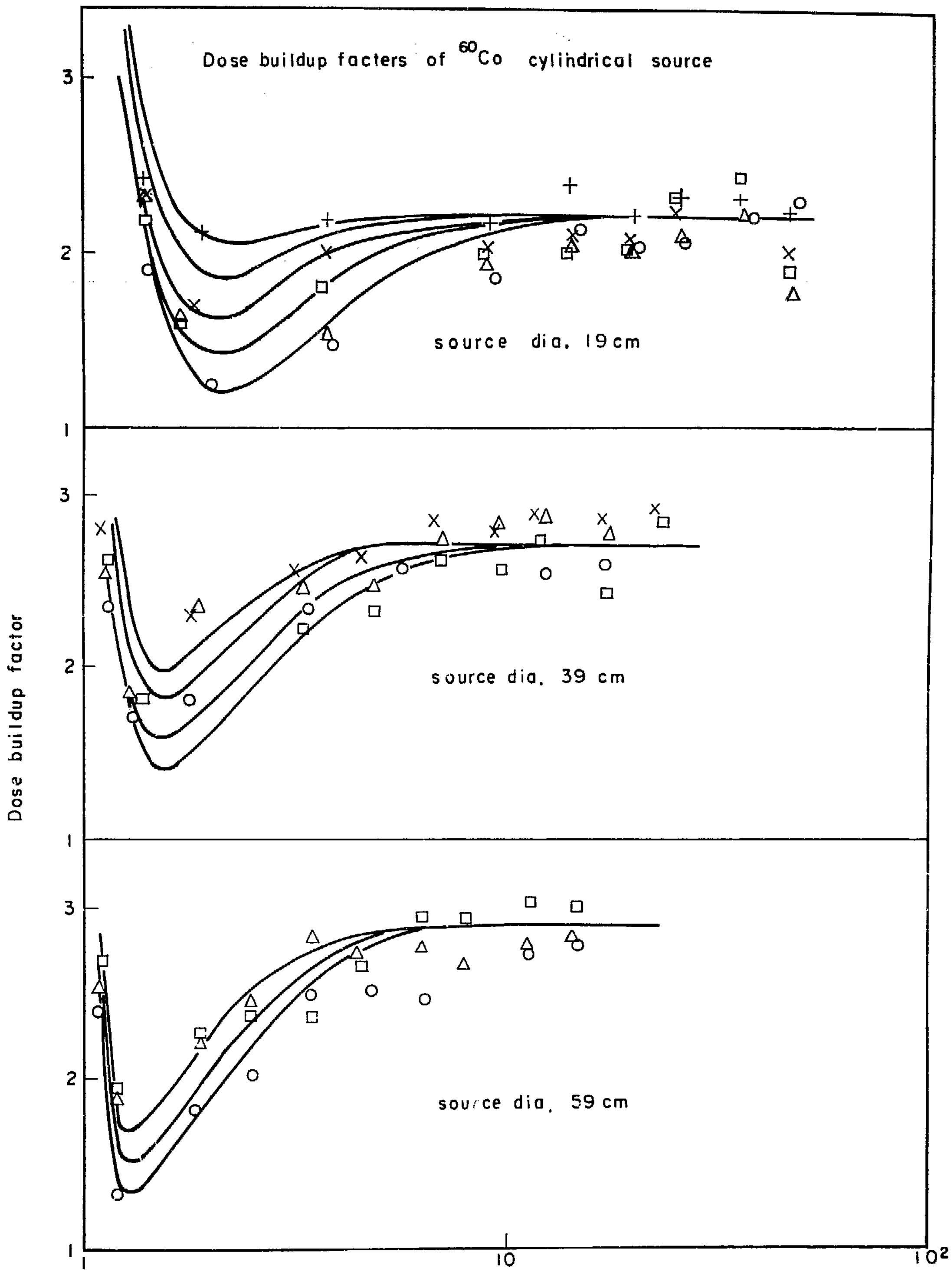


Fig. 12 Dose buildup factor of Double Layers



○ source hight 20 cm
△ 〃 40 cm
□ 〃 60 cm
× 〃 80 cm
+ 〃 100 cm

Fig. 13 Dose buildup factors of ^{60}Co cylindrical sources

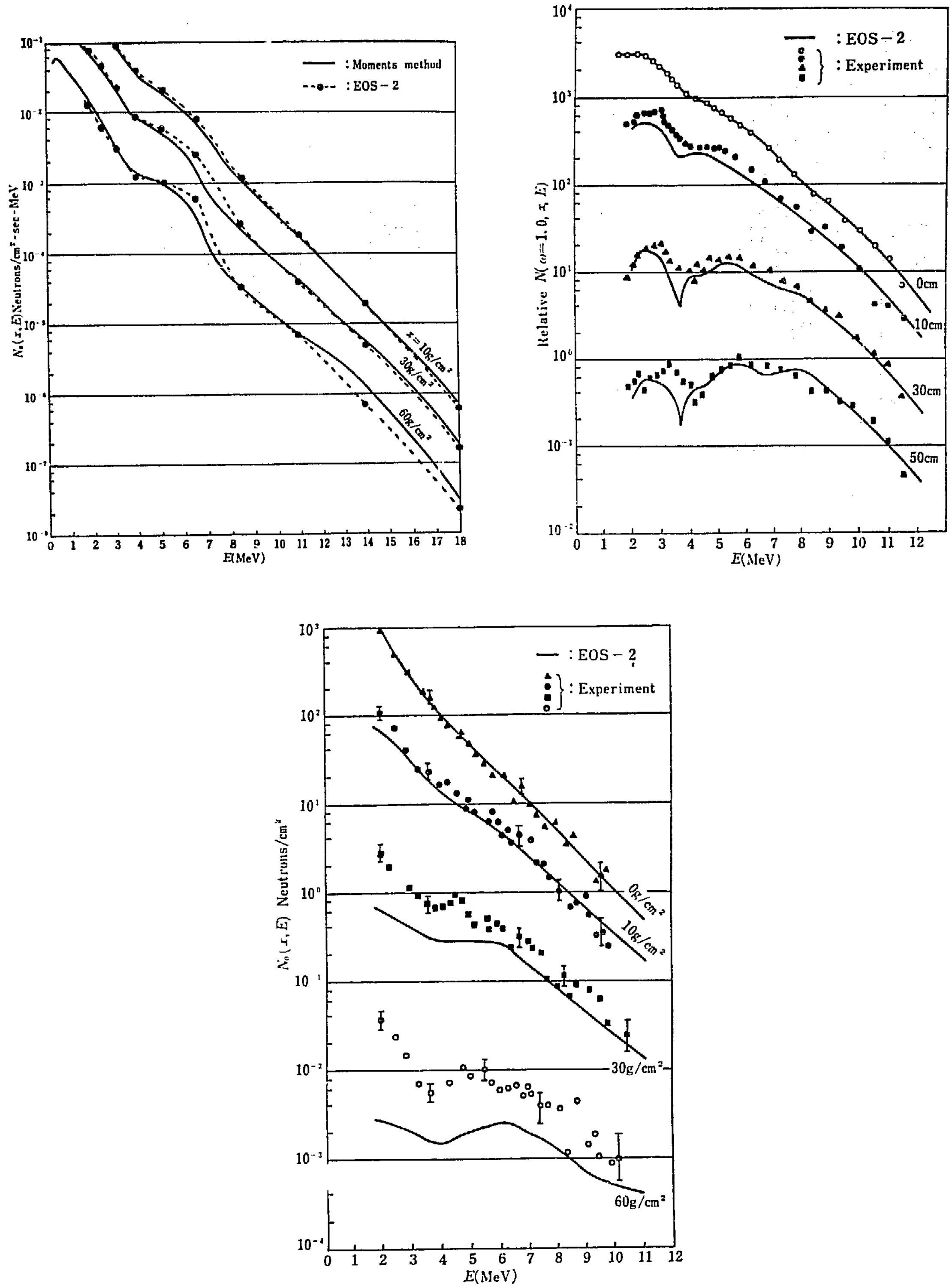


Fig. 14 Comparison of EOS-2 with moments method and experimental results

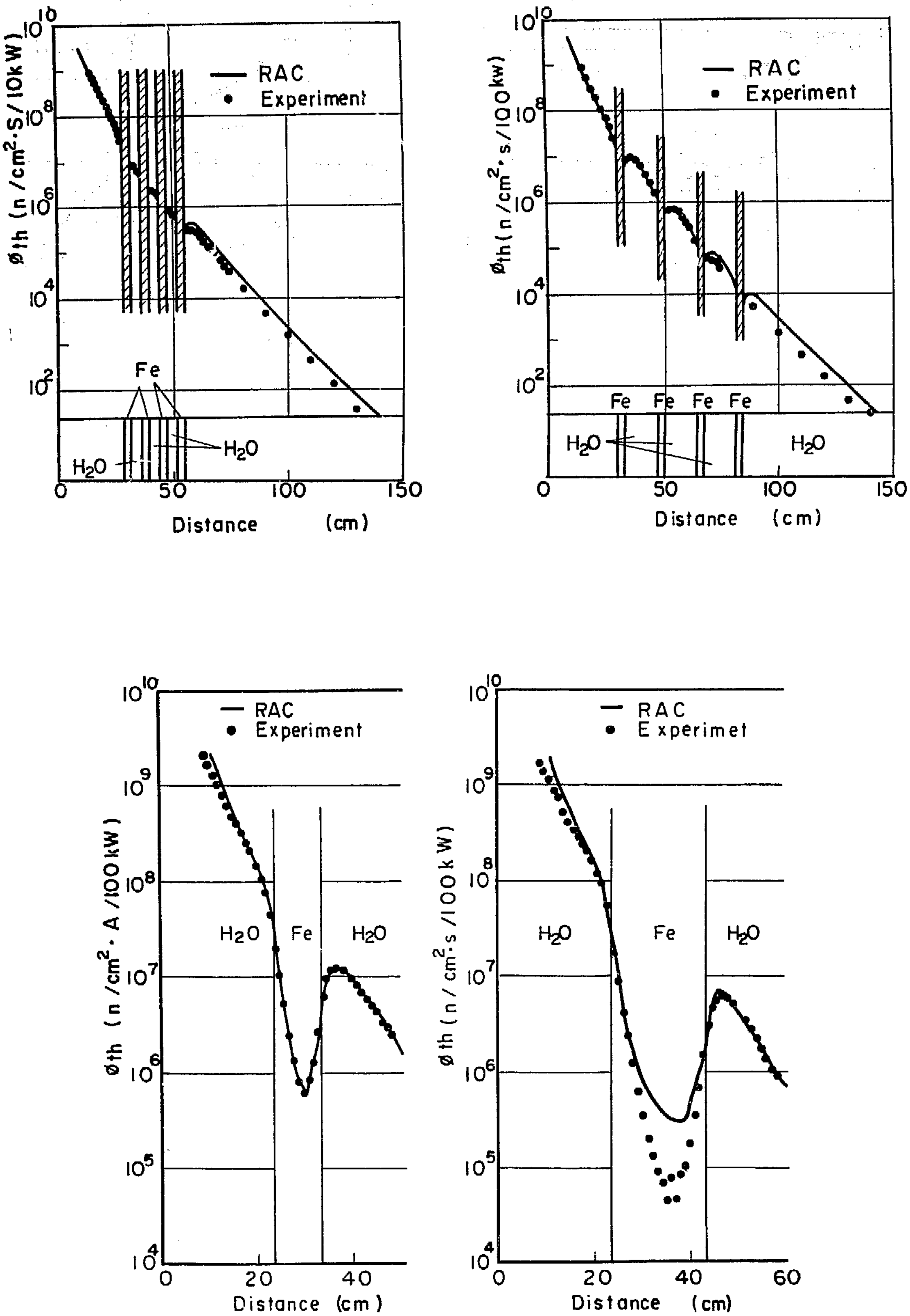


Fig. 15 Comparison of experimental results with calculation by RAC code

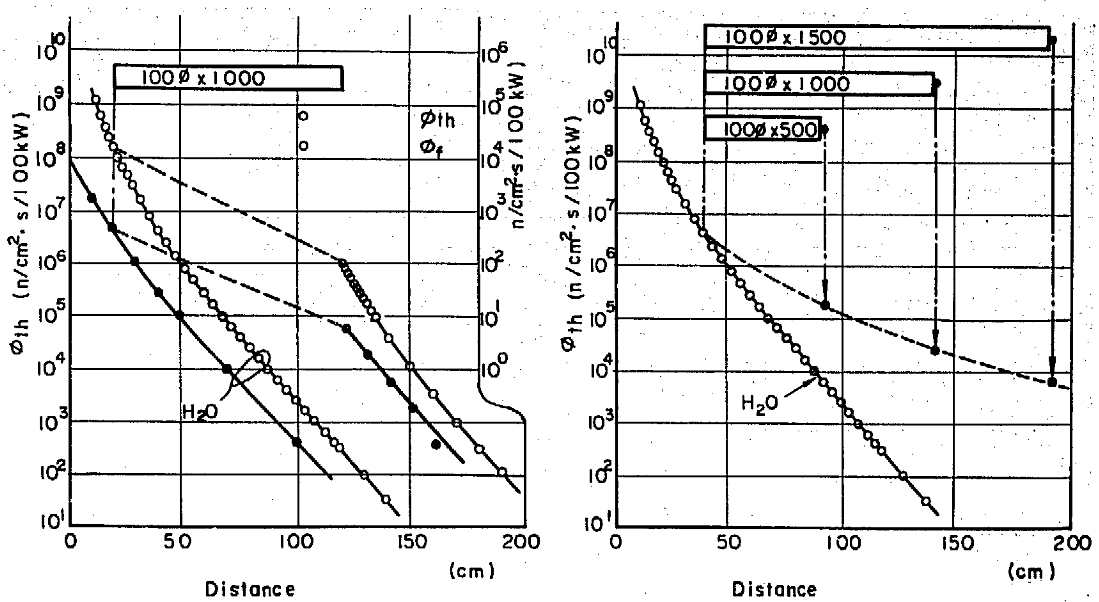


Fig. 16 Neutron streaming through acrylate pipe

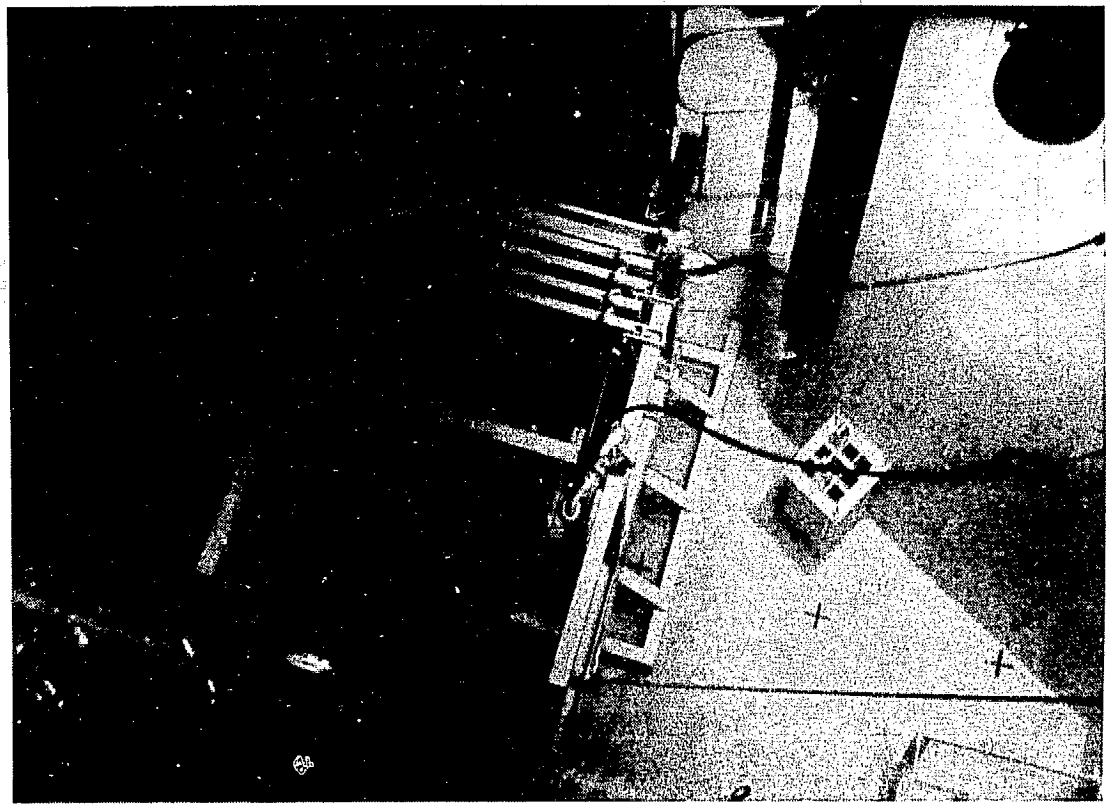


Photo 1 Iron multilayers in the No. 1 pool of JRR-4

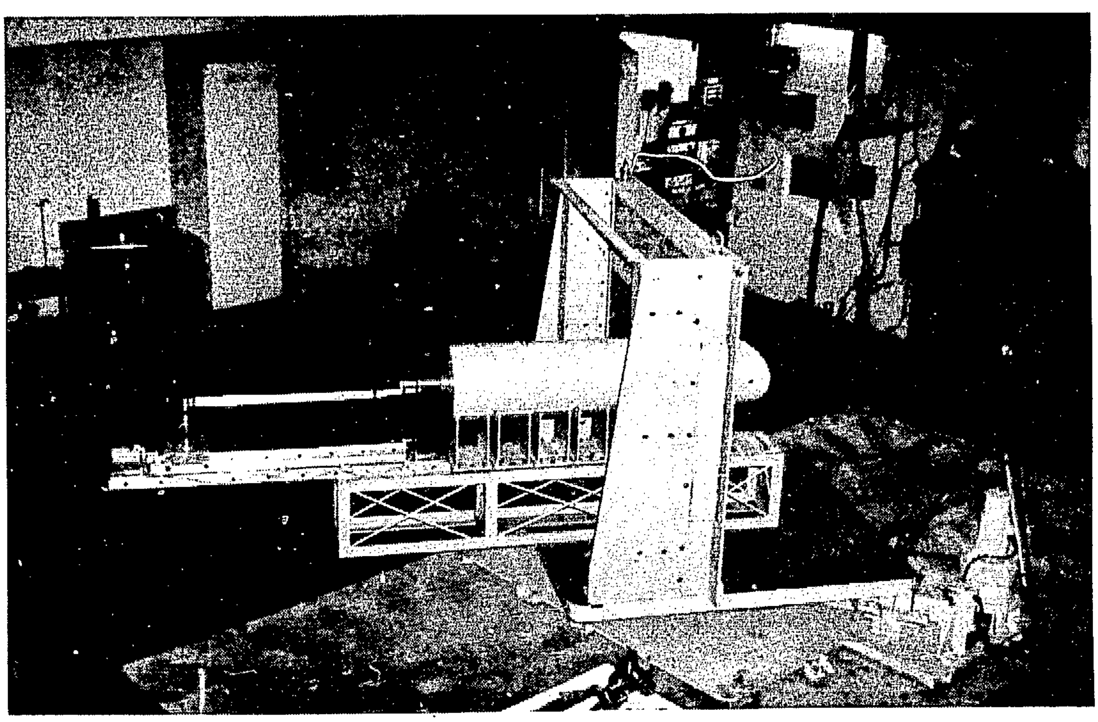


Photo 2 Arrangement of lead duct mockup in the dry shielding test facility of JRR-4