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Neutron radiation protection shielding — Design principles and considerations for the choice of appropriate materials

Écrans de protection neutronique — Principes de conception et éléments pour le choix de matériaux appropriés

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Foreword

ISO (the International Organization for Standardization) is a worldwide federation of national standards bodies (ISO member bodies). The work of preparing International Standards is normally carried out through ISO technical committees. Each member body interested in a subject for which a technical committee has been established has the right to be represented on that committee. International organizations, governmental and non-governmental, in liaison with ISO, also take part in the work. ISO collaborates closely with the International Electrotechnical Commission (IEC) on all matters of electrotechnical standardization.

International Standards are drafted in accordance with the rules given in the ISO/IEC Directives, Part 3.

The main task of technical committees is to prepare International Standards. Draft International Standards adopted by the technical committees are circulated to the member bodies for voting. Publication as an International Standard requires approval by at least 75 % of the member bodies casting a vote.

Attention is drawn to the possibility that some of the elements of this International Standard may be the subject of patent rights. ISO shall not be held responsible for identifying any or all such patent rights.

ISO 14152 was prepared by Technical Committee ISO/TC 85, *Nuclear energy*, Subcommittee SC 2, *Radiation protection*.

Annexes A to F of this International Standard are for information only.

Introduction

In its Publication 60 (1991), the International Commission on Radiological Protection (ICRP) recommended annual limits in terms of effective dose and equivalent dose to the skin, to the lenses of eyes the and to extremities. Due to the tissue weighting factors in its definitions, the effective dose is destined to be an instable quantity. In fact, its values strongly depend on the geometrical conditions of irradiation and can only be determined using anthropomorphic phantoms approaching the human body.

For monitoring purposes, the International Commission on Radiation Units and Measurements (ICRU) and the ICRP recommended so-called operational quantities [ICRU Report 39 (1985), ICRU Report 51 (1993) and ICRP Publication 74 (1996)] [4]. The operational quantity recommended for area monitoring of strongly penetrating radiation is the ambient dose equivalent.

Unfortunately the ambient dose equivalent does not give a conservative estimate of the effective dose for all the energies of interest for neutron radiation [ICRP Publication 74 (1996)] [4] and, at high energies, both for neutron and photon radiation. Moreover, the ambient dose equivalent has not been introduced for shielding purposes.

All the experimental data found in literature are expressed in terms of maximum dose equivalent, a quantity based on superseded Q vs LET relationship [ICRP Publication 21 (1971)] [3].

This International Standard makes use of the ambient dose equivalent as radiation protection quantity, based on the new Q-L relation as recommended in ICRP Publication 60. Due to the above-mentioned limitation, the field of application is restricted to neutrons of energy below 20 MeV.

For practical purposes, some data expressed in terms of maximum dose equivalent are also included in this International Standard. It is recommended that these data be conservatively multiplied by a factor 2 if used for purposes of shielding calculations [ICRU Publication 51 (1993)].

To establish an operational neutron radiation protection shielding system, several consecutive steps should be performed:

- choice of the radiological objectives and other design criteria;
- characterization of the radiation sources;
- identification of the constraints on placement and construction;
- choice of the shielding materials and arrangement within the shielding;
- implementation of a calculation method;
- choice of the final solution;
- experimental verification.

This International Standard outlines, in the body of the text, the basic requirements and the general provisions for the implementation of each of these previous steps.

Detailed annexes complete these general considerations, especially in the following fields:

- characterization of neutron sources;
- criteria for the choice of neutron shielding materials;
- review of shielding calculation methods (manual methods or computer codes).

NOTE The general principles provided in this International Standard are mainly applicable to simple slab geometries. For neutron shielding of complex geometries, including local lessenings such as service penetrations, ducts, labyrinths, zigzag mountings, the guidance given in this International Standard apply, but are generally not sufficient. For that purpose, special calculation techniques and arrangement of the shielding materials are needed. These specific considerations are out of the scope of this International Standard.

The same remark can be made for criticality shielding, for which the principles given in this International Standard apply, but are generally not sufficient. In this case additional considerations should be taken into account.

For these two items (criticality shielding and neutron radiation protection shielding of complex geometries), further standards will be studied in the near future.

Neutron radiation protection shielding — Design principles and considerations for the choice of appropriate materials

1 Scope

This International Standard presents the general methodology governing the design of neutron radiation protection shielding and the choice of neutron radiation protection shielding materials.

This International Standard is applicable to facilities and operations where neutron sources are located and used, and where workers are occupationally exposed. These operations and facilities vary considerably in design and purpose. These facilities and operations include, but are not limited to:

- nuclear power plants;
- research reactors;
- particle accelerators and neutron generators;
- fusion research facilities;
- transportation packaging for radioactive materials operations;
- medical treatment and research facilities and applications;
- industrial applications such as use of devices for measuring and detecting moisture and density level;
- space applications;
- calibration facilities;
- radiographic installations;
- nuclear fuel cycle installations (reprocessing plants, plutonium solution handling facilities, shielded cells, waste storage, etc.).

The criteria for the design of neutron shielding and the choice of shielding materials contained in this International Standard should be applied to the design of neutron radiation protection shielding systems in such facilities.

2 Normative references

The following normative documents contain provisions which, through reference in this text, constitute provisions of this International Standard. For dated references, subsequent amendments to, or revisions of, any of these publications do not apply. However, parties to agreements based on this International Standard are encouraged to investigate the possibility of applying the most recent editions of the normative documents indicated below. For undated references, the latest edition of the normative document referred to applies. Members of ISO and IEC maintain registers of currently valid International Standards.

ISO 921:1997, *Nuclear energy — Vocabulary*

ISO 7212, *Enclosures for protection against ionizing radiation — Lead shielding units for 50 mm and 100 mm thick walls*

ISO 8529-1, *Reference neutron radiations — Part 1: Characteristics and methods of production*

ISO 8529-3, *Reference neutron radiations — Part 3: Calibration of area and personal dosimeters and determination of response as a function of energy and angle of incidence*

ISO 9404-1, *Enclosures for protection against ionizing radiation — Lead shielding units for 150 mm, 200 mm and 250 mm thick walls — Part 1: Chevron units of 150 mm and 200 mm thickness*

ISO 15080, *Nuclear facilities — Ventilation penetrations for shielded enclosures*

ICRP 60 (1991), *1990 Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60, Annals of the ICRP, 21, (1-3), Pergamon Press, Oxford (1991)

ICRU 39 (1985), *Determination of Dose Equivalents Resulting from External Radiation Sources*, ICRU Report 39, 1985 (International Commission on Radiation Units and Measurements, Bethesda, MD)

ICRU 51 (1993), *Quantities and Units in Radiation Protection Dosimetry*, ICRU Report 51, 1993 (International Commission on Radiation Units and Measurements, Bethesda, MD)

ICRU 57 (1998), *Conversion Coefficients for use in Radiological Protection against External Radiation*, ICRU Report 57, 1998 (International Commission on Radiation Units and Measurements, Bethesda, MD)

ICRU 60 (1998), *Fundamental Quantities and Units for Ionizing Radiation*, ICRU Report 60, 1998 (International Commission on Radiation Units and Measurements, Bethesda, MD)

IAEA1) (1986), Safety Guides, Safety Series No. 76, *Radiation Protection Glossary*, Vienna

IAEA (1996), Safety Guides, Safety Series No. 115, *International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*, Vienna

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1) International Atomic Energy Agency.

3 Terms and definitions

For the purposes of this International Standard, the following terms and definitions apply.

3.1 Quantities [2] and units

3.1.1

equivalent dose in a tissue or organ

H_T

absorbed dose in a tissue or organ T, weighted for the type of radiation R

$$H_T = W_R \times D_{T,R}$$

where $D_{T,R}$ is the absorbed dose, averaged over tissue or organ T, due to the radiation R and W_R is the radiation weighting factor, which is dependent of the energy of the neutrons

[ICRP 60:1991, IAEA Safety Series No.115:1996]

NOTE 1 In the case of several radiation qualities, the equivalent dose in a tissue or organ, H_T, is the sum of the products of the radiation weighting factor W_R and the mean absorbed dose, $D_{T,R}$:

$$H_T = \sum_R W_R \times D_{T,R}$$

NOTE 2 The SI unit of the equivalent dose is the joule per kilogram ($J \cdot kg^{-1}$), with the special name sievert (Sv).

3.1.2

effective dose

E

sum of the weighted equivalent doses in all the tissues and organs of the body

$$E = \sum_T W_T \times H_T$$

where H_T is the equivalent dose in tissue or organ T and W_T is the weighting factor for tissue T

[ICRP 60: 1991, ICRU 51:1993, IAEA Safety Series No.115:1996]

NOTE 1 These tissue weighting factors take into account the varying sensitivity of organs with respect to stochastic effects of radiation. The weighting factor, W_T, represents the part of the contribution of the tissue T to the total exposure of the whole body when uniformly exposed ($\sum W_T = 1$).

The effective dose can also be expressed as the sum of the doubly weighted absorbed dose in all the tissues and organs of the body. The following applies based on the definition of H_T:

$$E = \sum_R \sum_T W_T \times W_R \times D_{T,R}$$

NOTE 2 The SI unit of the effective dose is the joule per kilogram ($J \cdot kg^{-1}$), with the special name sievert (Sv).

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[2] According to ICRP Publication 74, “equivalent dose” and “effective dose” are “Radiation Protection Quantities”, while “ambient dose equivalent”, e.g. is an “Operational Quantity”.

3.1.3
dose equivalent

H

product of Q and D at a point in tissue

$$H = D \times Q$$

where D is the absorbed dose and Q the quality factor at this point

[ICRU 51:1993]

NOTE The SI unit of equivalent dose is the joule per kilogram (J·kg^{-1}), with the special name sievert (Sv).

3.1.4
ambient dose equivalent

$H^*(d)$

dose equivalent, at a point in a radiation field, that would be produced by the corresponding aligned and expanded radiation field in the ICRU sphere at a depth d, on the radius opposing the direction of the aligned field

[ICRP 60:1991, ICRU 39:1985, ICRU 51:1993, IAEA Safety Series No.115:1996]

NOTE 1 A depth d = 10 mm is recommended for strongly penetrating radiation.

NOTE 2 The SI unit of ambient equivalent dose is the joule per kilogram (J·kg^{-1}), with the special name sievert (Sv).

NOTE 3 In the aligned and expanded radiation field, the fluence and the energy distribution have the same value in all the interested volume as at the measurement point in the real field; the field is called aligned.

3.1.5
ambient dose equivalent rate

$\dot{H}^*(d)$

quotient of d$H^*(d)$ per dt:

$$\dot{H}^*(d) = \frac{\mathrm{d}H^*(d)}{\mathrm{d}t}$$

where d$H^*(d)$ is the variation of the ambient dose equivalent $H^*(d)$ in the time interval dt

[ICRP 60:1991, IAEA Safety Series No. 76:1986]

NOTE The SI unit of ambient dose equivalent rate is the joule per kilogram and per second (J·kg^{-1}·s^{-1}), with the special name sievert per second (Sv·s^{-1}). A frequently used unit is sievert per hour (Sv·h^{-1}).

3.2 Other definitions

3.2.1
neutron fluence

Φ

quotient of dN by da:

$$\Phi = \frac{\mathrm{d}N}{\mathrm{d}a}$$

where dN is the number of neutrons incident on a sphere of cross-sectional area da

[ICRU 51:1993, ISO 8529-1]

NOTE The SI unit of the neutron fluence is m^{-2}, a frequently used unit is cm^{-2}.

3.2.2
neutron fluence rate
neutron flux density

φ

quotient of dΦ by dt:

$$\varphi = \frac{\mathrm{d}\Phi}{\mathrm{d}t} = \frac{\mathrm{d}^2 N}{\mathrm{d}a \times \mathrm{d}t}$$

where dΦ is the increment of neutron fluence in the time interval dt

[ICRU 51:1993, ISO 8529-1]

NOTE The SI unit of the neutron fluence rate is $\mathrm{m}^{-2}\cdot\mathrm{s}^{-1}$.

3.2.3
spectral neutron fluence
energy distribution of the neutron fluence

Φ_E

quotient of dΦ by dE:

$$\Phi_E = \frac{\mathrm{d}\Phi}{\mathrm{d}E}$$

where dΦ is the increment of neutron fluence in the energy interval between E and $E + \mathrm{d}E$

[ICRU 51:1993, ISO 8529-1]

NOTE The SI unit of the spectral neutron fluence is $\mathrm{m}^{-2}\cdot\mathrm{J}^{-1}$; a frequently used unit is $\mathrm{cm}^{-2}\cdot\mathrm{eV}^{-1}$.

3.2.4
spectral neutron fluence rate
spectral neutron flux density

φ_E

quotient of dΦ_E by dt:

$$\varphi_E = \frac{\mathrm{d}\Phi_E}{\mathrm{d}t} = \frac{\mathrm{d}^2\Phi}{\mathrm{d}E \times \mathrm{d}t}$$

where dΦ_E is the increment of spectral neutron fluence in the time interval dt

[ICRU 51:1993, ISO 8529-1, IAEA Safety Series No. 76:1986]

NOTE The SI unit of the spectral neutron fluence rate is $\mathrm{m}^{-2}\cdot\mathrm{s}^{-1}\cdot\mathrm{J}^{-1}$; a frequently used unit is $\mathrm{cm}^{-2}\cdot\mathrm{s}^{-1}\cdot\mathrm{eV}^{-1}$.

3.2.5
dose

measure of the radiation received or “absorbed” by a target

NOTE The quantities termed absorbed dose, organ dose, equivalent dose, effective dose, committed equivalent dose or committed effective dose are used depending on the context. The modifying terms are often omitted when they are not necessary for defining the quantity of interest.

[IAEA Safety Series No.115:1996]

3.2.6
dose limit

value of the effective dose or the equivalent dose to individuals from controlled practices that shall not be exceeded

[IAEA Safety Series No.115:1996]

3.2.7
dose equivalent

quantity used by the International Commission on Radiation Units and Measurements (ICRU) in defining the operational quantities ambient dose equivalent, directional dose equivalent and personal dose equivalent

NOTE The quantity dose equivalent has been superseded for radiation protection purposes by equivalent dose.

[IAEA Safety Series No.115:1996]

3.2.8
neutron fluence-to-dose equivalent conversion coefficient

h_Φ

quotient of the neutron dose equivalent, H, and the neutron fluence, Φ, at a point in the radiation field, undisturbed by the irradiated object:

$$h_\Phi = \frac{H}{\Phi}$$

[ICRU 57:1998, ICRU 60:1998, ISO 8529-1]

NOTE 1 The SI unit of the neutron fluence-to-dose equivalent conversion coefficient is $\mathrm{Sv \cdot m^2}$.

NOTE 2 Any statement of a fluence-to-dose equivalent conversion coefficient requires the statement of the type of dose equivalent, e.g. ambient dose equivalent, personal dose equivalent. The conversion coefficient, h_Φ, depends on the energy spectra and the directional distribution of the incident radiation. Their specific definitions and the respective values of the conversion coefficients are given in ISO 8529-3.

3.2.9
neutron source strength of a neutron source at a given time

B

quotient of $\mathrm{d}N^*$ by $\mathrm{d}t$:

$$B = \frac{\mathrm{d}N^*}{\mathrm{d}t}$$

where $\mathrm{d}N^*$ is the expectation value of the number of neutrons emitted by the source in the time interval $\mathrm{d}t$

[ISO 8529-1]

NOTE 1 The SI unit of the neutron source strength is s^{-1}.

NOTE 2 In this International Standard, when the term “intensity of a neutron source” is used, it is equivalent to the term “neutron source strength”.

3.2.10
spectral neutron source strength
energy distribution of neutron source strength

B_E

quotient of dB by dE:

$$B_E = \frac{\mathrm{d}B}{\mathrm{d}E}$$

where dB is the increment of neutron source strength in the energy interval between E and E + dE

[ISO 8529-1]

NOTE The SI unit of the spectral neutron source strength is $\mathrm{s}^{-1} \cdot \mathrm{J}^{-1}$; a frequently used unit is $\mathrm{cm}^{-2} \cdot \mathrm{s}^{-1} \cdot \mathrm{eV}^{-1}$.

3.2.11
neutron flux

$\dot{N}$

quotient of dN by dt:

$$\dot{N} = \frac{\mathrm{d}N}{\mathrm{d}t}$$

where *dN* is the increment of the number of neutrons in the time interval dt

[IAEA Safety Series No. 76:1986]

NOTE The SI unit of the neutron flux is s^{-1}.

3.2.12
kerma

K

quotient of dE_{tr} by dm:

$$K = \frac{\mathrm{d}E_{\mathrm{tr}}}{\mathrm{d}m}$$

where dE_{tr} is the sum of the initial kinetic energies of all the charged ionizing particles liberated by uncharged ionizing particles in a material of mass dm

[ISO 921]

NOTE 1 The SI unit of kerma is the joule per kilogram ($\mathrm{J} \cdot \mathrm{kg}^{-1}$), with the special name gray (Gy).

NOTE 2 The name kerma is derived from kinetic energy released in matter.

3.2.13
controlled area

area in which specific protection measures and safety provisions are or could be required for:

a) controlling normal exposures or preventing the spread of contamination during normal working conditions;

b) preventing or limiting the extent of potential exposures.

[IAEA Safety Series No.115:1996]

3.2.14

supervised area

any area not designated as a controlled area but for which occupational exposure conditions are kept under review even though specific protective measures and safety provisions are not normally needed

[IAEA Safety Series No. 115:1996]

3.2.15

exposure

act or condition of being subject to irradiation

NOTE Exposure can be either external exposure (irradiation by sources outside of the body) or internal exposure (irradiation by sources inside the body). Exposure can be classified as either normal exposure or potential exposure, either occupational, medical or public exposure and, in intervention situations, either emergency exposure or chronic exposure.

[IAEA Safety Series No. 115:1996]

3.2.16

shield

material interposed between a source of radiation and persons, or equipment or other objects, in order to attenuate the radiation

[IAEA Safety Series No. 76:1996]

3.2.17

shielding

action to calculate, design, select appropriate material, and verify the efficiency of a shield, in order to achieve the required functions with regard to the radiation protection objectives

3.2.18

source

anything (apparatus, substance, installation) that may cause radiation exposure, such as by emitting ionizing radiation or releasing radioactive substances or materials

[IAEA Safety Series No. 115:1996]

3.2.19

cross-section

probability of an interaction of a specified type between an incident particle, or an incident radiation, and a target particle, which permit the evaluation of the number of interactions between a particle flux, or radiation, and a target particles system

[IAEA Safety Series No. 76:1986, ISO 921:1997]

NOTE The SI unit of the cross-section is m^2, the symbol denoting cross section is σ. A frequently used unit is the barn: 1 barn = 10^{-28} m^2.

3.2.20

safety assessment

review of the aspects of design and operation of a source, which are relevant to the protection of persons or the safety of the source, including the analysis of the safety and protection provision established in the design and operation of the source and the analysis of risks associated with normal conditions and accident situations

[IAEA Safety Series No.115:1996]

4 Basic rules and principles for the definition of neutron shielding

4.1 Introduction

The purpose of neutron shielding is to protect workers and public from the ionizing radiation produced by neutron sources. Neutron emission is always associated with the production of gamma radiation in fission and fusion and some accelerator-produced reactions. Most neutron interactions within the shielding material also produce gamma radiation. Shielding should be designed to protect occupationally exposed workers and the general public from neutron and gamma radiations.

Gamma shielding unit design as defined in ISO 7212 and ISO 9404-1 may be integrated in the neutron shielding design.

4.2 Basic principles

The basic principles of a neutron shielding design for neutrons are the same as those used in the development of gamma radiation shielding. The use of time, distance and shielding apply, albeit differently. The overall approach reduces the personal doses to workers and the collective doses to general public. In review, taking into account the ALARA principle (see 5.5), these principles are applied as follows:

- Time of exposure: the time that areas adjacent to the shielding are occupied (time available for exposure) should be carefully determined and minimized.
- Distance: the distance between the exposed person and the neutron source should be optimized (maximized).
- Shielding: must be provided to first moderate the neutron, second absorb as far as possible the residual neutron flux and third minimize the gamma radiation originating from the neutron absorption process. Doses due to neutrons depend on the neutron fluence and its spectral distribution.

NOTE In certain special uses, and in addition to the previous requirements, the neutron source strengths can be reduced in order to allow intervention close to the radiation sources. It is the case of experimental reactors where the “neutron source” can be driven and adapted to the required level.

In this International Standard radiological objectives are expressed in terms of ambient dose equivalent, $H^*(10)$, which is considered to give a conservative estimate of effective dose.

When used without modifying quantity, the terms “dose” and “dose rate” will mean ambient dose equivalent and ambient dose equivalent rate respectively.

The quantities used in making the calculation are the particle fluence and the energy distribution of the particle fluence. The results are converted into ambient dose equivalent using a set of energy dependent conversion coefficients.

4.3 Methodology

To determine an operational neutron radiation protection shielding system, an iterative process shall be implemented, taking into account the following steps, considered in the given order:

1) Choice of design criteria — choice of dose objectives and other design criteria.
2) Characterization of the radiation sources.
3) Identification of constraints on placement and construction.
4) Choice of the shielding materials and arrangement within the shield.
5) Choice of the calculation methods.
6) Choice of the final solution.
7) Experimental verification.

The general architecture of this iterative process is given in Figure 1.

These steps are discussed in detail in clauses 5 to 11. It is further recommended that they be addressed in the same order. The decisions made as a result of considering these criteria should be documented and retained as permanent facility records and used as a basis for design control and for periodical safety reviews.

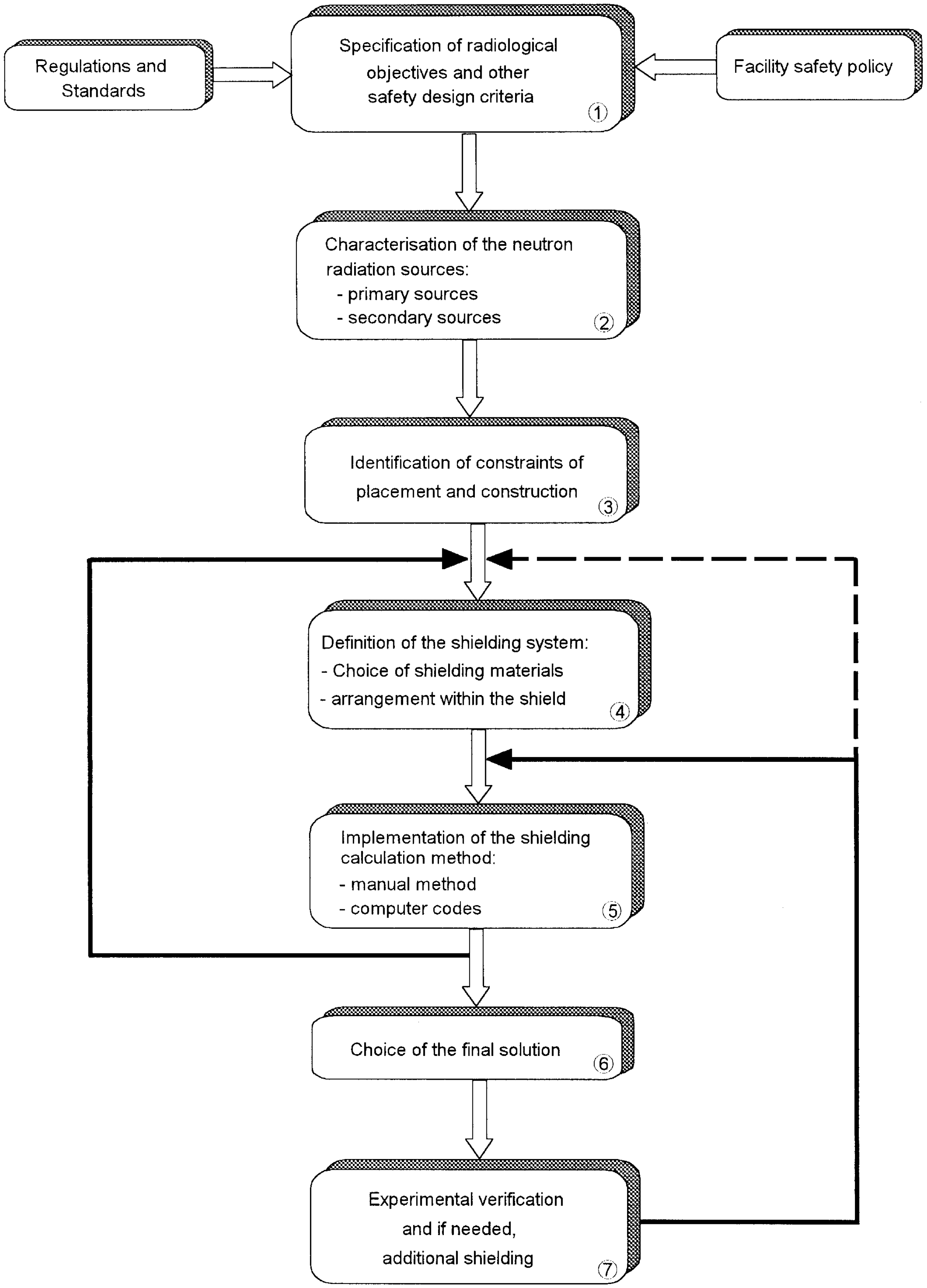


Figure 1 — General architecture of the methodology used to establish an operational neutron radiation protection shielding system

5 Choice of radiological objectives and other design criteria

5.1 General

The purpose of this step is to identify all the factors which have to be considered during the dimensioning of the neutron shielding; at first the radiological objectives in terms of ambient dose and ambient dose rate limits and at second all the additional architectural, mechanical, chemical and safety objectives that the shielding has to satisfy, according to the general purpose of the nuclear plant or the operation.

The first criterion, consists of determining the maximal ambient dose equivalent and ambient dose equivalent rate at the different working areas and on the factory boundary, with respect to the radiation protection regulation concerning the exposure of workers and general public to ionizing radiation. In accordance with ICRP recommendations and according to the ALARA principle, these values should be kept as far below the regulatory limits as is reasonably achievable (see 5.4). Additional considerations for the selection of these radiological objectives are given in 5.2 to 5.4.

The second series of criteria concerns all the additional safety requirements with which the shielding shall comply with respect to the general purpose of the plant or the installation (e.g. participation to the containment wall, playing a role of gamma radiation shielding, participation to the fire compartment, need for decontamination, necessity of removable parts). In order to achieve complete implementation, all kinds of hazards or constraints such as mechanical constraints, chemical constraints, fire constraints, thermal constraints, architectural constraints, general safety constraints, shall, in addition, be considered.

Consequently, persons involved in shielding design should review the locally applicable radiation protection regulations and requirements as well as regulations concerning building, fire protection, factory safety or other regulations applicable at the facility or for the operation. Examples of items to be considered are discussed in 5.2.

5.2 Classification of work areas (based upon general occupancy, limited occupancy, etc. and on specific protection requirements)

5.2.1 General

The control of radiation sources is helped by requiring that the workplaces containing exposure risk be formally designated. The ICRP recommendations use two such official designations: controlled areas and supervised areas. For practical reasons, some local regulations, have introduced a third category of designation, which is called unrestricted areas.

5.2.2 Controlled areas and supervised areas

5.2.2.1 A controlled area is one in which normal working conditions, including the possible occurrence of minor mishaps, require the workers to follow well-established procedures and practices aimed specifically at controlling radiation exposure.

5.2.2.2 A supervised area is one in which the working conditions are kept under review but special procedures are normally not required.

5.2.3 Unrestricted areas

An unrestricted area is a working area where it is not necessary to take any special health precautions. An unrestricted area may be inside or outside a building or in part of a building, and shall be characterized by unlimited access and egress by the general public.

5.2.4 Dose limits within these workplaces

NOTE 1 The dividing line between controlled areas and supervised areas, if the latter are used, has commonly been set with the aim of ensuring that the doses to workers in the supervised areas can confidently be predicted to be less than 3/10 of the occupational dose limits. ICRP now regards this definition as being too arbitrary and recommends, in its publication 60 (1991), that the designation of controlled and supervised areas be decided either at the design stage or locally by the operating management on the basis of operational experience and judgement. This judgement has to take into account the expected level on the likely variations of the doses and intakes, and the potential for accidents.

NOTE 2 As a consequence of the previous designations, an unrestricted area could be considered as an area where doses do not exceed 1 mSv per year or a dose corresponding to 1/10 of the annual occupational dose limits fixed for workers exposed to radiation and for whom it is not necessary to foresee any particular measures to afford protection from radiation.

National competent authorities shall elaborate criteria or orientation for the classification of areas taking into account specific circumstances. The requirements stated in these national regulations shall be selected prior to receiving the ICRP recommendations.

5.3 Determination of the occupancy and frequency of work

The total effective dose in a controlled or supervised area is the summation of the effective doses received as a result of tasks conducted within the area. The effective dose for each single task is the product of the effective dose rate and the exposure time. It is important for the shielding system designer to separate:

- persons for whom the shielded area is a permanent work place and for which the shielding is being designed;
- persons who have to work for a short duration with some frequency of occurrence (recurrence, cumulative total duration, etc.);

and establish exposure times and exposure frequency for each.

5.4 Choice of design criteria in terms of radiological objectives

5.4.1 Specification for dose and dose rate limitation

For the purposes of this International Standard, the definitions of ICRU Report 51 and ICRP 60 shall be applied to dosimetry terminology. The system for dose regulation recommended in this International Standard is based on recommendations of the International Commission on Radiological Protection (ICRP 60). Users of this International Standard should familiarize themselves with the applicable national or international systems for dose limitation in use at the facility or in effect for the operation prior to selecting shielding materials and designing the protection shield.

The recommended dose limits are stated in terms of the effective dose and the equivalent doses to the lenses of the eyes, the skin and the extremities.

The radiological objectives are usually expressed as a percentage of the applicable occupational dose limit or the dose limit for members of the general public.

- The currently recommended occupational dose limitation of the ICRP as stated in ICRP Report 60 (1991) is 20 mSv effective dose per year, averaged over five years (100 mSv in 5 a), with the further provision that the effective dose should not exceed 50 mSv in any single year. The 5-year period shall be defined by the regulatory authority, e.g. as discrete 5-year calendar periods. The annual equivalent dose to the lenses of the eyes, the skin and the extremities are respectively 150 mSv, 500 mSv and 500 mSv. Therefore the protection shield shall ensure that the ambient dose equivalent $H^*(10)$ does not exceed the previous values.
- According to the same ICRP recommendations, the dose limit for a member of the general public is 1 mSv effective dose per year. Higher dose limits can be exceptionally authorized, but in no case shall the averaged value over 5 years exceed 1 mSv a^{-1}.

Where dose limitation systems are stated in national standards or regulations, the corresponding system supersedes the ICRP recommendations listed above.

5.4.2 Dose from mixed radiation sources and/or internal contamination

The effective total dose, resulting from combined external exposure to neutrons and gamma radiation and/or from internal contamination due to inhaled or ingested contaminants shall be considered in the establishment of the dose objectives.

5.4.3 Derived dose equivalent rate limits for shielding calculation

From a practical point of view, for determining the thickness of a protection shield against external exposure, it is useful to consider, instead of the previous ambient dose limits, the ambient dose equivalent rate limits obtained from these values, which take into account the exposure time.

E.g. these can be:

— an average of 400 µSv·week^{-1},

— an average of 10 µSv·h^{-1}, for a 40-hour working week (or 2 000 h·a^{-1}).

5.5 Application of the ALARA principle

For practical purposes, the dose constraint, which is a fractional quantity of the individual dose limit for radiation workers, shall be selected in accordance with the ALARA (As Low As Reasonably Achievable) principle. The ALARA principle specifies that exposure to ionizing radiation shall be kept as far below regulatory limits as is reasonably achievable – social, environmental and economic impacts being taken into account.

5.6 Normal, off-normal and emergency operating requirements

Operating and emergency specifications shall be prepared to support normal, off-normal and emergency operating conditions. These specifications should address, as a minimum, the following subjects.

a) Establishment of controlled, supervized and restricted areas.

b) Control of personnel access into the controlled, supervized and restricted areas.

c) Establishment of stay times under normal, off-normal and emergency conditions.

d) Requirements for the use of personal protective clothing, respiratory protective equipment, etc., for controlled, supervized and restricted areas.

e) Maximal doses and dose rates for normal, off-normal and emergency conditions.

f) Controls for inadvertent criticality for the sources of fissile material used or stored within the installation or under the protection of shielding materials.

5.7 Other items to be considered

Considerations which should be taken into account in designing neutron shielding and selecting the shielding material include, but are not restricted to, the items listed below.

a) Inventory of other sources of neutron and gamma radiation exposure which can contribute to the total dose.

b) Examination of possible increases in source strengths as a result of normal operations, or emergency situations associated with accidents.

c) Possibility of internal exposure due to inhalation of airborne radioactivity or ingestion of surface contamination.

d) Dose and dose rate objectives for the work location.

e) Dose and dose rate objectives at contact with the face of the protection shielding, if needed.

f) Dose and dose rate objectives at the boundary of the working or storage area.

g) Dose and dose rate objectives for the facility boundary.

h) The shielding geometry and the contribution of radiation leakages.

i) The contribution of “skyshine” or scattered component of radiations to the dose and dose rate objectives at the working place and at the facility boundary.

5.8 Special criteria or safety considerations

Numerous other factors may be considered, all of which may have impact on either the architectural or engineering design of the neutron shielding. Criteria such as those shown in a) to e) shall be considered.

a) The potential contributions of gamma and other particles to the dose objective.

b) The effect or impact of introduced or surrounding objects (such as water tanks, containment walls, etc.).

c) Compliance with safety code regulations.

d) Resistance to radiation.

e) Resistance to fire, to pressure, to corrosion, to shocks, etc.

NOTE When water tanks or natural protection shields constitute a part of the protection against radiation, care should be taken to keep these in operation during all stages of the life of the installation.

6 Characterization of neutron sources

6.1 Characteristics of the neutron sources

Neutron sources are characterized by:

- a reaction that produces neutrons;
- source strength, which is the number of neutrons emitted per unit of time or specific source strength, which is the specific source strength is the number of neutrons emitted per unit of time per unit of mass;
- neutron energy or neutron energy spectrum;
- time structure of neutron radiation: this concerns, e.g., accelerators;
- contributions of other kinds of radiation.

The principal neutron source strengths and typical neutron spectra are described in annex A. Annex B presents the basic neutrons producing reactions.

6.2 Description of the source medium

The source medium is a specific part of the problem. Each source must be characterized by its:

- geometry (point source, linear, cylindrical spherical source or complex geometry);
- physical characteristics (bare or cladded source, equipment of different geometric form, surface deposit or residues of radioactive particles, density);

- physical state (solid, liquid or gas);
- chemical and isotopic composition and their variation with time (ageing, radioactive decay and increase of activity due to decay products);
- energy and angular source strength distribution.

6.3 Nature of the source

6.3.1 General

Two kinds of radiation are emitted and are addressed in most shielding problems, these are neutrons and gamma radiations. Sources of radiation that shall be addressed are primary or direct sources of radiation and secondary sources of radiation that emit neutrons and/or gamma radiation as a consequence of reacting with the radiation emitted by the primary sources of radiation.

6.3.2 Primary sources

These emit direct gamma as well as neutron radiation, e.g.:

- nuclides that decay by spontaneous fission;
- actinide solutions, which emit spontaneous neutrons due to (α, n) reactions;
- the core of a nuclear reactors;
- fresh or irradiated fuel assemblies;
- the plasma of a fusion reactor;
- deposits or residues of radioactive products in glove boxes, primary ducts, filters or ventilation ducts;
- the mixed material matrix of a neutron source such as Pu-Be sources;
- the targets for an accelerator.

6.3.3 Secondary sources or induced sources

These emit neutrons, gamma radiation and other radiations as a consequence of interaction of the radiations emitted by the primary source with the structural materials or the materials constituting the shielding wall. These sources include the following:

- production of secondary gamma radiation due to the interaction of neutrons with the shielding materials;
- various materials located in the exposure room or in the reactor vessels, etc., that can become activated when submitted to neutron radiation. This includes structural frameworks, containment materials, shielding, walls, coolants and atmosphere of the rooms;
- radiation created by neutron multiplication (induced fission);
- used accelerator targets.

6.4 Relative importance of sources with mixed radiation fields

In the case of a mixed (neutron and gamma) field, which is the general case since gamma radiation is generally more or less associated with neutrons, the following procedure may be followed:

- If the neutron and gamma radiation sources are of equal importance (similar contribution to the ambient dose equivalent rate), the neutron and gamma protection shielding will be defined for both sources simultaneously. The resulting shielding may be homogeneous or heterogeneous or a combination of materials.
- If one of the radiation sources is much more important than the other, the protection shielding will be defined on the basis of the most important source, including scattered component of radiation, while ensuring that the resulting shielding is also sufficient for the less important source. If not, the design shall be modified.

7 Identification of constraints on placement and construction

7.1 General

In the design of neutron shielding all the normal and accident constraints and load conditions to which the protection materials may be submitted shall be examined. In case of a highly specific use, additional criteria and qualification tests may be required, or some of the criteria listed in 5.8 may apply; e.g., highly portable sources of neutrons used for construction or radiography apparatus shall withstand the rigors of transportation and be resistant to external conditions, rather than will be the case in a permanent installation.

In addition, the following thermal, electrical, mechanical, physical, structural, architectural, chemical, safety, and economic constraints and resistance to radiation, as defined in 7.2 to 7.10, should be considered as appropriate.

7.2 Thermal constraints

As appropriate, the shielding materials as well as their surface protection or casing, shall be able to withstand the environmental temperatures associated with its location as well as any heat induced while performing its design function. The protection shield shall e.g.:

- retain its tensile and other physical characteristics in its design environment;
- be fire resistant (reaction to fire);
- not add to the fire load at equal neutron shielding performance;
- dissipate heat easily or be easily cooled;
- not overheat as a result of nuclear interactions;
- retain intermolecular water.

7.3 Electrical constraints

The shielding materials shall not be subject to a build-up of free charge as a consequence of performing its shielding function and shall possess adequate electrical earthing.

7.4 Workability and other mechanical constraints

As appropriate, the shielding materials shall possess good material characteristics that permit it to be easily worked and shaped. The shielding material shall be resistant to the normal wear and tear of the work place and to natural phenomena such as tornados, earthquakes, floods, etc., which potentially could occur during the lifetime of the shielding. The characteristics which could be required are e.g.:

- resistance to deformation or elongation;
- workability;
- resistance to impact;
- resistance to wear, crumbling, flaking;
- smooth, hard surfaces which are easily decontaminated;
- resistance to seismic action.

7.5 Physical constraints

The following physical characteristics (or physical constraints) of the shielding materials shall be available (or specified):

a) mass;

b) density;

c) physical state (solid or liquid);

d) composition;

e) watertightness (resistance to the passage of water);

f) surface conditioning.

7.6 Installation, structural and architectural constraints

The shielding design must fit into the facility. Fit is determined by the following factors:

a) permissible floor loading;

b) available space including elevation;

c) need for fixed or portable installation;

d) ease of assembly and disassembly;

e) necessity of vision through the shielding material;

f) available routes for the movement of materials, availability of lifts;

g) requirements for support or self-supporting.

In addition, the shielding materials shall not constitute a potential source of aggression for the external environment (e.g. missiles, in case of shocks).

7.7 Chemical constraints

The shielding materials shall withstand the chemicals in the operating environment including cleaning and decontamination agents. This shall include e.g.:

a) resistance to corrosion (acids, bases, solvents, salt mist, etc.);

b) resistance to thermal or oxidation ageing;

c) resistance to water;

d) resistance to decontamination agents and cleaners;

e) resistance to toxic vapours;

f) resistance to visible and ultra violet light induced reactions;

g) resistance to the build-up of activation products (the shielding material shall be relatively free of trace elements).

In addition, the shielding materials shall not constitute a potential source of aggression for the external environment (e.g. emission of toxic substances, liquefaction).

7.8 Resistance to radiation

The shielding materials shall be able to withstand radiation during their time in service and conserve their efficiency against gamma radiation and neutrons. When required, the resistance to radiation of the shielding material shall be improved.

7.9 Constraints imposed by other safety considerations

As appropriate, the shielding design shall also address constraints imposed by safety requirements such as fire resistance, prevention of criticality.

7.10 Economic constraints

The shielding design shall address certain economic and environmental issues such as the potential for re-use, cost of construction and disposal, and material availability.

If required, appropriate regulatory requirements concerning the issues of neutron shielding at the end of their life time (waste disposal, re-use), protection of the environment, shall be considered. These constraints shall be taken into account during the design stage, when shielding materials are selected. Corresponding obligations can e.g. concern decontamination behaviour, exemption of activation products and use of relevant surface protection.

In addition, the ALARA principle shall be followed in the management of the economic constraints encountered during the development of the design of the shielding system.

8 Choice of materials

8.1 Recommended types of material

8.1.1 General

Materials used for the design of neutron radiation protection shielding shall have large cross-sections for neutron moderation and capture. These factors permit the shielding to perform its two primary functions, moderate the energy of the incident neutrons and reduce the neutron fluence. Three types of material, which may be used singly or combined in a composite shielding, are recommended for protection against the different types of radiation. They are described in 8.1.2 to 8.1.4.

8.1.2 Hydrogenous materials

Low-density materials, containing hydrogen, will slow down, i.e., reduce the energy of the incident neutrons. The material with the highest possible hydrogen content shall be selected. The greater the number of hydrogen atoms present per cubic centimetre, the greater the moderating capability of the material. The following are particularly suitable:

- water;
- hydrocarbon chain materials (such as polyethylene, polypropylene, etc.);
- paraffin;
- compressed wood;
- plasters;
- compounds of other mineral powders;
- hydrides;
- concretes.

8.1.3 Neutron absorbing materials

When the neutrons have been moderated or have been slowed down, they can more easily react with atoms in the shielding or be captured. Gamma radiations are produced by these reactions. The most common suitable materials, those with high reaction cross sections for thermal neutron capture (2 200 m·s^{-1} or 0,025 3 eV) which shall be used in conjunction with hydrogenated materials are:

- boron 10 [σ_a (n, α) ≈ 4 000 barn – σ_a (n, γ) ≈ 0,1 barn] — boron produces 558,8 keV and 657,3 keV gamma radiation;
- natural boron [σ_a (n, α) ≈ 800 barn];
- cadmium 113 [σ_a (n, γ) ≈ 20 000 barn] — cadmium produces high energy gamma radiation in the range 4 MeV to 7 MeV;
- natural cadmium [σ_a (n, γ) ≈ 2 500 barn];
- water [σ_a (n, γ) ≈ 4 barn] — hydrogen (in water) produces gamma radiation of 2,2 MeV.

NOTE 1 The cross sections depend on the energies of the incident neutrons. The values indicated above correspond to an energy of 0,025 3 eV and are extracted from [6].

NOTE 2 When selecting the neutron absorbing material, care should be taken over the energy of the resulting gamma radiation. Generally a compromise between the efficiency of the neutron capture and the quantity and energy of the secondary gamma produced, shall be chosen.

NOTE 3 Boron also emits an energetic alpha particle following neutron capture. Since the range of alpha particles is short, the resultant energy deposition following alpha slowing down can give rise to a very significant heat generation problem (e.g. when control rods are introduced in the core of advanced gas-cooled reactors, in order to shut down the power rating).

8.1.4 High density materials

Some high density materials, such as iron, shall be added to the neutron shielding because they help moderate the fast neutrons by the inelastic scattering mechanism, which prevails for neutrons with energies greater than 1 MeV. High density materials also attenuate gamma radiation.

Care should be exercised when using material with $Z > 50$ because they have significant cross sections (n, 2n or greater) for reactions which could increase the neutron fluence. Annex B presents the energy thresholds for some of the (n, 2n) and other nuclear reactions.

8.2 Relative efficiency of materials

Various shielding materials have different attenuating characteristics for neutrons and gamma radiation because of the differences between neutron scattering and capture cross sections.

Depending on the relative importance of these two types of emission, as demonstrated by their relative contribution to the dose objective or to the dose rate objective without the shielding, shielding materials shall be selected to ensure that the resultant dose or dose rate, neutron plus gamma, is lower than the design specified dose objective.

The following properties shall be considered in making the choice of shielding materials.

- In general, materials containing light nuclei are more efficient in attenuating neutron energy.
- Materials containing heavy nuclei are more efficient in attenuating gamma radiation.

Typical material specifications that specify the number or hydrogen atoms contained in the material as well as the percentage of certain absorbers are listed in annex C.

As an example, the curves shown in Figure 2 show the relative neutron and gamma contribution to the total dose for a few commonly used materials according to the type of neutron source.

8.3 Arrangement of materials in the shielding

8.3.1 General

Source characteristics may have an impact on the order by which material is placed in the shielding.

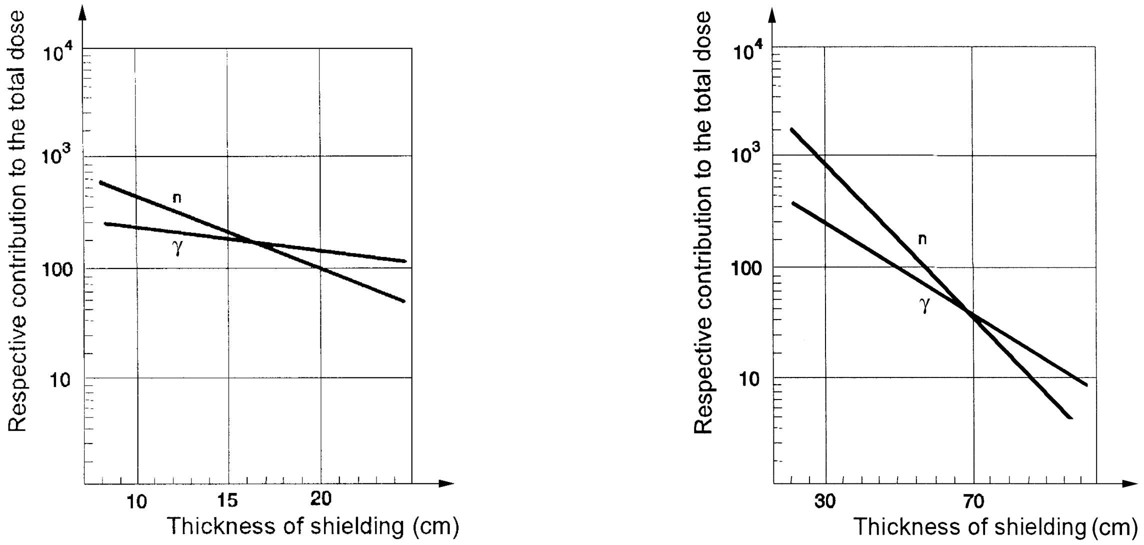
When there are layers of different materials in a shielding, it may be important to arrange them in an order that takes advantage of their attenuation characteristics.

8.3.2 Neutron shielding for fast neutron sources

For a fast neutron source, it is particularly recommended to install moderating materials as the first part of the shield in order to bring the neutron beam rapidly (after a few impacts) to the thermal or epithermal state. The recommended moderating materials are compounds rich in hydrogen, carbon or oxygen.

High density materials, such as iron or cast iron, used for gamma shielding or for structural materials with a large inelastic scattering cross section, also contribute to degrading the energy of fast neutrons.

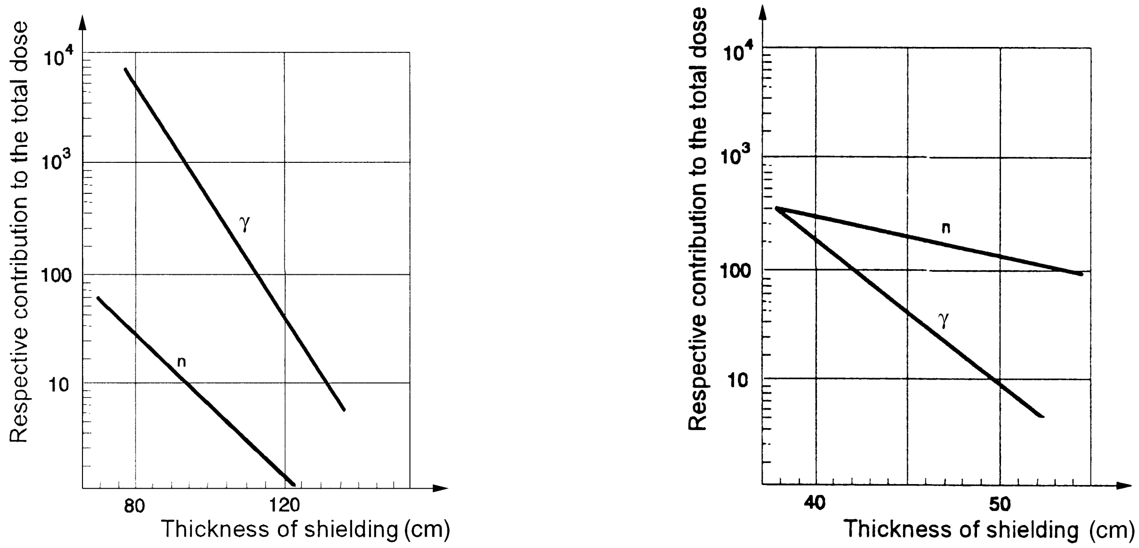
Following the moderating material, or possibly integrated in the same compound as the moderating material shall be materials with a large capture cross section for thermal neutrons. A material containing boron is ideal for this purpose.



Boron polyethylene shielding

Ordinary concrete shielding

a) Case of a source spectrum constituted by irradiated PuO_2, after 6 a decay



Ordinary concrete shielding

Cast iron shielding

b) Case of a source spectrum constituted by vitrified waste fission products

NOTE In the present calculation, the considered sources are spherical sources, of respective diameter of 14 cm [case a)] and 20 cm [case b)], while the shields are of plane geometry, placed at 20 cm from the source.

Figure 2 — Relative compared neutron and gamma contribution to the total dose, for some usual materials

8.3.3 Slow and thermal neutrons

Materials with a large capture cross section for thermal neutrons shall be selected as the prime component of the shielding. A material containing boron, which has the advantage of generating low energy capture gamma radiation (0,5 MeV), is ideal for this purpose.

8.3.4 Neutron and gamma mixed sources

It is usual to design the shielding in order to attenuate a mixed field of neutrons plus gamma radiation as in the case of a nuclear reactor. For this case, the shielding material used can be high density concrete, which has good moderating characteristics as well as providing good gamma radiation shielding.

For other cases, it may be desirable to place a layer of high Z material before the moderator to absorb the gamma radiation and moderate the neutrons through inelastic scattering. That may be reinforced with a layer of material with good moderating characteristics with an outer layer of material with high neutron capture cross sections.

8.3.5 Arrangement of shielding material

The arrangement of the shielding material shall take into account the following considerations.

- If the source has a large proportion of gamma radiation, the largest part of the gamma radiation should be absorbed in the first layer of the shielding. The following layers, containing neutron absorbers, are thus protected against gamma radiation damage and will have a longer life span.
- In case of cylindrical geometry (transportation cask), the material of highest density shall be installed on the smallest diameter thus achieving a considerable saving in weight.
- When a large amount of capture gamma radiation is produced inside the shielding, protection can be achieved by placing high density layers (lead, depleted uranium, iron, etc.) on the outside of the shielding. Iron, however, will generate high energy capture gamma radiation in the presence of the remaining neutrons.
- The arrangement of the shielding layers or the composition of the shielding material can be redefined during the calculation process.

8.4 Special considerations for complex shielding including weak points

When weak points of the current shielding are unavoidable (service penetrations, ducts, labyrinths, zigzag mountings, local lessenings), additional shielding shall be placed on the radiation leakage lines, inside of the main shielding layer, outside of the shielding layer or directly included in the shielding layer.

For the design of the additional shielding, the following guidance shall be taken into account (see also ISO 15080). The additional shielding shall ensure that the shielding efficiency of the considered lessening, taken in all directions, provides the same level of the shielding as the current shielding layer.

The design of the additional shielding is dependent upon:

- the intensity and the position of the source (or sources) of radiation;
- the geometry of the weak points;
- the thickness of the current shielding layer.

The design of the additional shielding shall be conducted on a case by case basis. Care shall be taken to include in the total ambient dose equivalent the contribution of the scattered component of neutron sources through the shielding lessenings, which can rise to a very significant level compared to the direct contribution.

For the calculation of radiation leakages, the following special computer codes are appropriate: ALBATROS, NARCISSE, MERCURE, MCNP, TRIPOLI, MCBEND.

8.5 Choice of appropriate neutron shielding material

Annex C presents summarized tables giving the main characteristics of the most commonly used neutron shielding materials.

9 Choice of the calculation method

The choice of the method of calculation depends on the complexities faced in the shielding design. The design of simple shielding that is relatively uncomplicated shall be performed using simple straightforward methods. The more complex the designs, the more complex the method of analysis.

Manual calculation methods like those detailed in annexes E and F can be used for the definition of the preliminary design.

Detailed transport calculations shall be required for the development of the final design. Only those codes that are internationally recognized by shielding design specialists shall be used for final design calculations. Other codes shall only be used after their performance has been validated by comparison with reference codes that are internationally accepted or by comparison with experimental measurements.

Sometimes, the use of experimental models, in addition to theoretical calculation, can be helpful (see Note 2 of clause 11).

A list of computer codes that are available at the date of publication of this International Standard is given in annex D.

10 Choice of the final solution

Final shielding choice is usually a compromise taking into account environmental constraints and other safety considerations, the relative efficiency of the available shielding materials and their arrangement within the shielding, and cost constraints.

The process often involves successive steps or iterations:

- initial selection of shielding materials, after the constraints study;
- preliminary calculations on these series of materials;
- a second selection of shielding materials, in order to choose the final materials with which the final calculations are achieved and the final geometry established.

It should be noted however that no single method or solution is universal and no material is ideal. It is always the result of a compromise and depends on the order of priority of the criteria.

11 Experimental verification

After shielding has been installed, a check shall be made to ensure that it does indeed perform the required function, taking into account the different uncertainties (calculation, measurement). This check can be the following:

a) Compliance with code calculations: in this test, a comparison between the ambient dose equivalent rates behind the shielding and the specifications is achieved.

b) Verification at special points: this verification consists of the measurements of ambient dose equivalent rates at singular points behind the shielding which could not be simulated in the design calculations or on the verification of the homogeneity of the efficiency of the shielding materials which are not easily workable (lead or resin casting in a metallic casing of special profile, casting of concrete in cavities, etc.).

Should verification fail to establish that dose objectives have been met, additional shielding may be needed to provide adequate levels of protection.

NOTE 1 The previous tests are independent from the verifications tests implemented during the fabrication phases or the in-site mounting phases, which are done in compliance with the quality assurance procedures set by the project manager or the engineering companies in charge of the design of the shielding system.

NOTE 2 For certain special applications, namely for complex shielding geometry, experimental models reproducing partly or fully the real shielding system, are necessary to confirm the validation of the shielding design before the choice of the final solution. The use of these experimental models can allow, at the same time, the validation of the calculation methods used.

Annex A
(informative)

Considerations on neutron sources

A.1 Basic neutron producing reactions

A.1.1 (α, n) reactions

Some alpha-emitting radionuclides such as radium, radon, polonium and the transuranium isotopes can trigger (α, n) reactions in the light elements such as lithium, beryllium, boron, fluorine, oxygen or aluminium. Typical portable neutron sources are made of inter-metallic compounds of Pu, Am or Cm and Be.

The characteristics of the neutron source depend on the intensity and spectrum of the α source, the stopping power of the medium and the effective (α, n) reaction cross section of the neutron emitting nuclides.

An example of a neutron spectrum obtained by an (α, n) reaction is shown in Figure A.1.

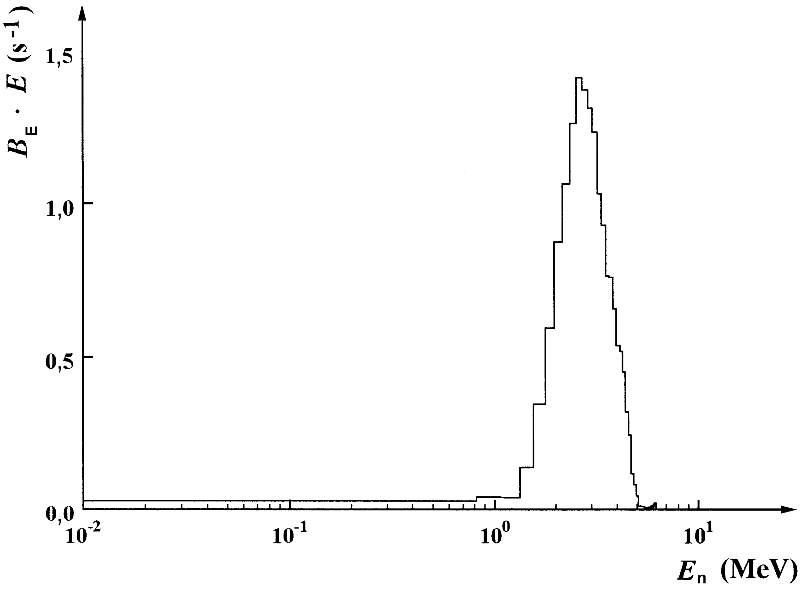


Figure A.1 — Neutron spectrum from an ^{241}Am-B(α, n) source

A.1.2 (γ, n) reactions

Some light elements have a loosely bound neutron, which can be emitted after photon excitation. The most common reactions are D(γ, n)H and ^{9}Be(γ, n)^{8}Be. These reactions may be very important in a fission reactor when heavy water or beryllium is used as moderator. Portable neutron sources are usually combinations of emitters like ^{24}Na, ^{56}Mn, ^{72}Ga, ^{88}Y, ^{116}In, ^{124}Sb, ^{226}Ra, ^{238}Pu or ^{239}Pu associated with Be, B, Li or D_2O.

The energy spectrum can be approximated to fission neutrons spectrum (see clause A.3).

A.1.3 Neutrons generated by accelerators

The production of neutrons with low energy accelerators results mostly from the nuclear reactions, (p, n), (d, n), and (α, n) between the accelerated particles or nuclei, alpha particles, and the atoms in the targets. When the accelerated particles are electrons, their deceleration in the target produces bremsstrahlung which triggers (γ, n) reactions.

The angular and energy distributions of the emitted neutrons are strongly dependent on the type and energy of the projectile as well as on the type of reaction.

A.1.4 Nuclear fusion

In thermonuclear fusion machines the main reactions are D(D, n)^{3}He and D(T, n)^{4}He. Therefore, fusion neutrons have a typical line spectrum. The neutron spectrum is peaked around 2,45 MeV and 14,1 MeV. The width of the peaks depends on the plasma temperature and on the possible additional heating on the plasma.

Figure A.2 shows the neutron spectrum obtained in fusion machines.

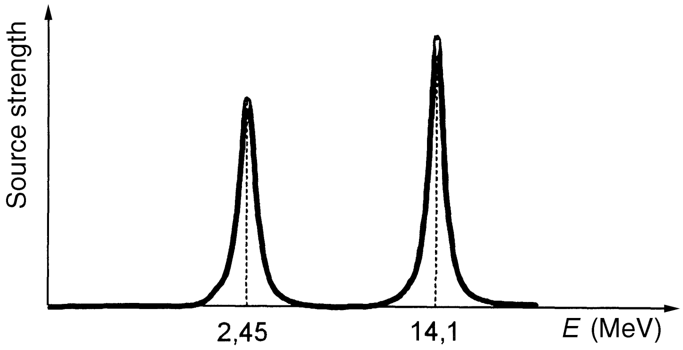


Figure A.2 — Neutron spectrum from neutron sources obtained in fusion machines

A.1.5 Nuclear fission

Fission neutron sources can be spontaneous or induced. The spectra of both fission sources are similar, but their intensities must be calculated differently. Primary sources, which emit direct radiation are listed below:

- actinide solutions;
- the core of a nuclear reactor;
- fresh or irradiated fuel assemblies.

Tables B.1 to B.4 in annex B show some of the neutron-producing reactions.

A.2 Neutron source strength

A.2.1 Fusion neutrons

The emission rate for fusion neutrons emitted by fusion machines is difficult to calculate. It depends on the density of deuterium and tritium atoms as well as on the temperature and density of the resultant plasma. For a high neutron yield the plasma temperature has to be 100 K or more.

D-D reactions have a threshold of about 1,2 MeV. An energy of 1,2 MeV is equivalent to $1{,}16 \times 10^{10}$ K. D-T reactions take place at about $1{,}7 \times 10^{9}$ K.

The production of fusion neutrons by neutron generators and other accelerators are described in A.2.2.

A.2.2 Radionuclide-based neutron sources

Source strength for commercial radionuclide-based neutron sources is usually quoted in terms of the activity of the radionuclide that produces the radiation that interacts to produce the neutron.

The most commonly used sources are ^{252}Cf, ^{241}Am-B(α, n) and ^{241}Am-Be(α, n). Their main characteristics are listed in Table A.1. The emitted neutron spectra are given in clause A.3.

Table A.1 — Reference radionuclides neutron sources

Source	Half-life a	Fluence average energy MeV	Dose equivalent average energy MeV	Specific source strength $s^{-1} \cdot kg^{-1}$	Ratio of "photon" to "neutron" dose equivalent rates	Spectrum averaged fluence-to-dose equivalent conversion coefficient $pSv \cdot cm^2$
$^{252}Cf - D_2O$ moderated	2,65	0,55	2,1	$2{,}1 \times 10^{15}$	0,18	105
^{252}Cf	2,65	3,13	2,3	$2{,}4 \times 10^{15}$	0,05	385
				$s^{-1} \cdot Bq^{-1}$		
^{241}Am-B(α, n)	432	2,72	2,8	$1{,}6 \times 10^{-5}$	$< 0{,}20$	408
^{241}Am-Be(α, n)	432	4,16	4,4	$6{,}6 \times 10^{-5}$	$< 0{,}05$	391

A.2.3 Spontaneous fission neutron sources

Some heavy actinide isotopes (with atomic numbers greater than 90) split spontaneously producing an average number (ν) of neutrons per spontaneous fission. The associated spontaneous fission half-lives are generally longer than those of radioactive decay through α or β emissions for these nuclides except in the case of some very heavy actinides, ^{248}Cm to ^{254}Cf, where they are of the same order of magnitude.

When there is no self-absorption, the source strength B and specific source strength B_s are given by the following formulae:

$$B = \frac{(\ln 2) \times \nu \times N}{T_{sp}} \text{ and } B_s = \frac{(\ln 2) \times \nu \times A}{T_{sp} \times M}$$

where

ν is the average number of neutrons per spontaneous fission;

N is the number of atoms in the source;

T_{sp} is the spontaneous fission half life, expressed in seconds;

A is Avogadro's number;

M is the atomic mass of the nuclide.

A.2.4 Accelerator produced neutrons

The source strength for accelerator produced neutrons depends on the number of incident particles per unit time, the number of target atoms, the reaction cross section (per type of atom at the energy of the incident particles) and the thickness of the target.

A.2.5 Induced fission neutron sources

In a sub-critical assembly, where $k_{\text{eff}} < 1$, the source strength B of the induced fission source, is represented by the following formula:

$$B = \frac{\sum_i b_i}{1 - k_{\text{eff}}}$$

where

b_i represents the source strengths of the different neutron sources described above;

k_{eff} is the effective multiplication factor of the assembly, representing the balance between production and consumption of neutrons.

A.3 Emission spectrum of neutron sources

A.3.1 Fusion neutrons

Nuclear fusion reactions produce neutrons with characteristic energies, 2,45 MeV for D-D reactions and 14,1 MeV for D-T reactions. Such fusion reactions can be produced in accelerator targets and fusion machines.

A.3.2 Spontaneous ^{252}Cf fission source

Figure A.3 shows the neutron energy spectrum of a spontaneous ^{252}Cf fission source (see ISO 8529-1).

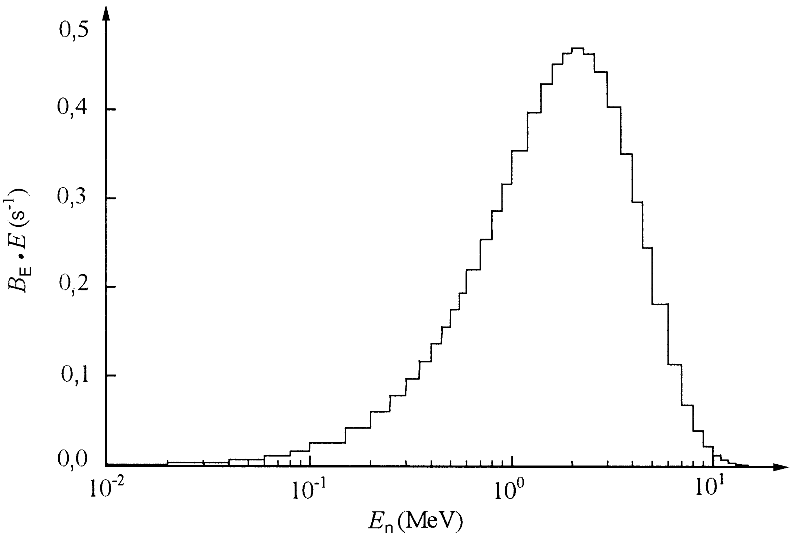


Figure A.3 — Neutron energy spectrum from a ^{252}Cf spontaneous fission source

A.3.3 Accelerator produced neutrons

The average neutron energies for selected accelerator produced neutrons are shown in Tables B.1 and B.2.

A.3.4 Fission neutrons

The emission spectrum of spontaneous or induced fission neutrons is well represented by the Watt–Cranberg formula shown below[3]:

$$N(E) = 2 \times \mathrm{e}^{-B/4A} \times A\sqrt{\frac{A}{\pi \times B}} \times \mathrm{e}^{-A \times E} \times \sinh\left(\sqrt{B \times E}\right)$$

where

$N(E)$ represents the neutron energy distribution function, expressed in MeV^{-1};

A and B are two parameters, expressed in MeV^{-1};

E is the energy, expressed in MeV.

For the fissile isotopes $^{235}\mathrm{U}$, $^{238}\mathrm{U}$, $^{239}\mathrm{Pu}$, the values of A and B are as follows:

	$^{235}\mathrm{U}$ (thermal fission)	$^{238}\mathrm{U}$ (fast neutron fission)	$^{239}\mathrm{Pu}$ (fast neutron fission)
A	1,036 3	1,02	1,02
B	2,29	2,12	2,495

Figure A.4 presents an example of such neutron energy distribution, calculated for a fission neutron source of $^{235}\mathrm{U}$.

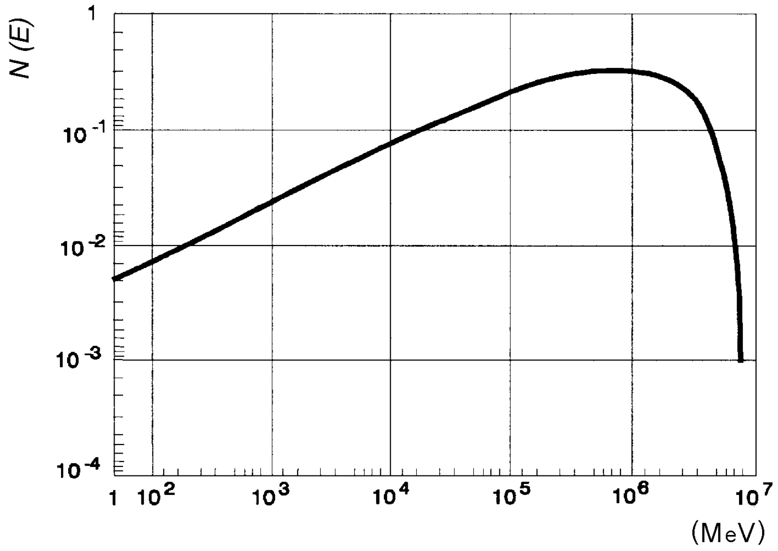


Figure A.4 — Neutron energy distribution for a typical fission neutron source ($^{235}\mathrm{U}$)

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[3] Reactor Handbook, Vol. III, part A, H. Soodak.

A.3.5 Neutron sources produced by (α, n) reactions

The reaction of an alpha particle with a light nucleus is employed to provide neutron sources for industrial applications. The average neutron energy for typical sources is shown in Tables B.1 and B.2. The most commonly used sources, whose spectra are given below, are:

- actinium–beryllium
- americium–beryllium
- curium–beryllium
- plutonium–beryllium
- radium–beryllium
- thorium–beryllium

The emission of neutrons is due to the reaction:

Be + $\alpha \rightarrow$ n + C (α being emitted by the heavy element associated with the beryllium).

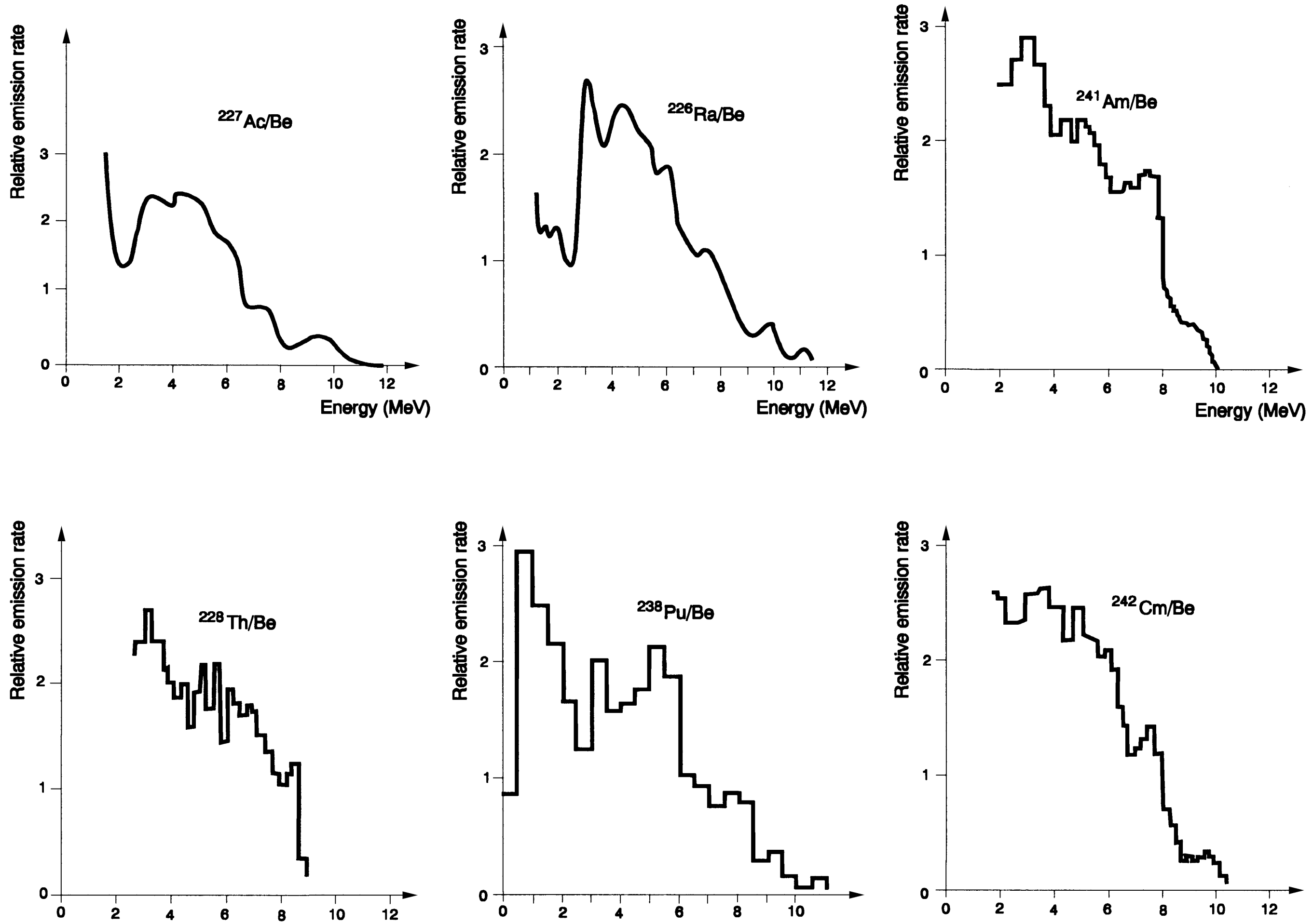

Figure A.5 — Energy spectra of some neutron sources produced by (α, n) reactions

A.4 Energy losses of the emitted neutrons

Collisions of the neutrons emitted by the source with the surrounding matter results in energy losses. Because the probability of the interactions of neutrons with matter (inelastic and elastic scattering, nuclear reactions and radiative captures) varies with their energy (energetic neutrons have longer free paths), the neutron spectrum will be degraded, and will differ from the emission spectrum.

The degrading of the energy spectrum is an important function of the neutron shielding. Reducing neutron energy significantly reduces the dose produced by a given unit of neutron fluence.

Annex B
(informative)

Neutron sources and threshold reactions

B.1 Main (γ, n) reactions

Table B.1 lists the characteristics of the main (γ, n) reactions.

Table B.1 — Characteristics of some important (γ, n) reaction sources

Source	Half-life (nuclide produced)	Principal photons		Average neutron energy [a] MeV	Standard yield [b]
		E MeV	Number per decay		
^{24}Na + Be	15,6 h	2,754	1,00	0,967	3,5
^{24}Na + D_2O	15,6 h	2,754	1,00	0,262	7,3
^{24}Al + Be	2,34 min	1,779	1,00	0,100	
^{56}Mn + Be	2,58 h	1,810	0,272	0,128	0,78
		2,113	0,143	0,397	
		2,522	0,010	0,761	
^{56}Mn + D_2O	2,58 h	2,522	0,010	0,146	0,08
		2,657	0,007	0,214	
^{72}Ga + Be	14,1 h	1,861	0,053	0,173	1,4
		2,202	0,259	0,746	
		2,491	0,077	0,733	
		2,508	0,128	0,748	
^{72}Ga + D_2O	14,1 h	2,491	0,077	0,131	1,6
		2,508	0,128	0,139	
^{76}As + Be	26,3 h	1,788	0,003	0,108	1,9
		2,096	0,007	0,382	
^{88}Y + Be	107 d	1,836	0,993	0,151	2,7
		2,734	0,006	0,949	
^{88}Y + D_2O	107 d	2.734	0,006	0,252	0,08
^{116m}In + Be	54,2 min	2,112	0,154	0,396	0,22
^{124}Sb + Be	60,2 d	1,691	0,490	0,022	5,1
		2,091	0,057	0,378	
^{140}La + Be	40,4 h	2,522	0,034	0,761	0,08
^{140}La + D_2O	40,4 h	2,522	0,034	0,146	0,2
^{226}Ra + Be	1622 a	many		0,68 max.	0,8
^{226}Ra + D_2O	1622 a	many		0,11 max.	0,03
^{228}Ra + Be	6,7 a	2,620		0,848	0,95
		1,800		0,119	
^{228}Ra + D_2O	6,7 a	2,620		0,195	2,6

NOTE Sources: Blizard and Abott [1962] and Shlein [1992].

[a] According to reference, assuming isotropic photoneutron emission.

[b] Number of neutrons emitted from 1 g of Be or D_2O per 10^6 disintegrations of the radionuclides placed 1 cm away. Yield is for all photons emitted.

B.2 Main (α, n) reactions

Table B.2. lists the characteristics of the main (α, n) reactions.

Table B.2 — Characteristics of some important (α, n) reaction sources

Source	Half-life	Principal alpha energies MeV			Average neutron energy MeV	Optimum neutron yield per 10^6 primary alpha [a]
^{210}Po/Li	138,4 d	3,305			0,48	1,3
^{239}Pu/Be	24 100 a	3,155;	5,143;	5,105	4,6	60
^{210}Po/Be	138,4 d	5,305			4,5	70
^{238}Pu/Be	87,8 a	5,499;	5,457;	5,358	4,5	80
^{241}Am/Be	432 a	5,486;	5,443;	5,388	4,4	75
^{244}Cm/Be	18,1 a	5,805;	5,763		4,3	100 [b]
^{242}Cm/Be	163 d	6,113;	6,070		4,1	110
^{226}Ra/Be	1 600 a	7,687;	6,003;	5,490	3,9	500 [c]
+ daughters		5,304;	4,785;	4,602		
^{227}Ac + Be	21,8 a	7,386;	6,819;	6,623	3,9	700 [c]
+ daughters		6,038;	5,960;	5,715		
^{241}Am/B	432 a	5,486;	5,443;	5,388	3	13
^{210}Po/C	138,4 d	5,305				0,10
^{241}Am/F	432 a	5,486;	5,443;	5,388	1,5	4,1
^{210}Po/F	138,4 d	5,305				5
^{210}Po/Na	138,4 d	5,305				1

NOTE Sources: Jaeger (1968), GPO (1970) and Knoll (1989).

[a] Yield for alpha particles incident on a target thicker than the alpha-particle range.

[b] Does not include a 4 % contribution from the spontaneous fission of ^{244}Cm.

[c] Yields are dependent on the proportion of daughters present. Value for ^{226}Ra corresponds to a 22 year old source (50 % contribution for ^{210}Po).

B.3 Neutron production by nuclear reaction involving light nuclei 4)

Table B.3 lists the characteristics of the neutrons produced by typical nuclear reactions involving light nuclei.

Table B.3 — Energy of neutrons produced by different nuclear reactions involving light nuclei

Target	Bombarding particle	Energy of bombarding particle MeV				
		0	1	2	5	10
		Neutron energy at 0° MeV				
12C	d	—	0,69	1,68	4,64	9,57
3H	p	—	—	1,20	4,22	9,23
7Li	p	—	—	0,23	3,33	8,35
13C	α	2,07	3,20	4,16	7,00	11,68
2H	d	2,45	4,14	5,24	8,24	13,02
9Be	α	5,27	6,68	7,71	10,60	15,23
3H	d	14,05	16,75	18,26	21,98	27,42

B.4 Fast neutron threshold reactions 5)

Table B.4 lists the main neutron threshold reactions.

Table B.4 — Some important neutron threshold reactions

Target	Reaction	Energy threshold MeV	E_n MeV	σ mb	Half-life (nuclide produced)
B	^{10}B (n, α) ^{7}Li	exoth [a]	th	3,84 × 10^6	stable
N	^{14}N (n, 2n) ^{13}N	11,31	(p, n) [b]	2,5	10,1 min
N	^{14}N (n, p) ^{14}C	exoth [a]	th	1 830	5 730 a
O	^{16}O (n, 2n) ^{15}O	16,65	(p, n) [b]	1,3	2,1 min
O	^{16}O (n, p) ^{16}N	10,24	(p, n) [b]	6,2	7,3 s
F	^{19}F (n, 2n) ^{18}F	12,10 [c]			110 min
Na	^{23}Na (n, γ) ^{24}Na	exoth [a]	th	530	15 h
Mg	^{24}Mg (n, p) ^{24}Na	7,00 [c] / 4,93 [a]	6	2	15 h
Mg	^{26}Mg (n, γ) ^{27}Mg	exoth [a]	th	38,2	9,46 min
Al	^{27}Al (n, γ) ^{28}Al	exoth [a]	th	231	2,3 min
Al	^{27}Al (n, p) ^{27}Mg	4,50 [c] / 1,90 [a]	5	25	9,46 min
Al	^{27}Al (n, α) ^{24}Na	7,10 [c] / 3,25 [a]	6	1,5	15 h

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4) ICRP publication 21 (1971).

5) From Neutron Contamination from Medical Electron Accelerators, National Council on Radiation Protection and Measurements, Report No. 79, 1984.

Table B.4 (*continued*)

Target	Reaction	Energy threshold MeV	E_n MeV	σ mb	Half-life (nuclide produced)
Si	^{28}Si (n, p) ^{28}Al	4,00 [a]	5	20	2,3 min
P	^{31}P (n, γ) ^{32}P	exoth [a]	th	172	14,3 d
P	^{31}P (n, p) ^{31}Si	2,40 [c] / 0,73 [a]	2	20	2,62 h
S	^{32}S (n, p) ^{32}P	2,70 [c] / 0,96 [a]	2	18	14,3 d
Ar	^{40}Ar (n, γ) ^{41}Ar	exoth [a]	th	660	110 min
Ti	^{46}Ti (n, p) ^{46}Sc	3,80 [c]			88,3 d
Ca	^{48}Ca (n, γ) ^{49}Ca	exoth [a]	th	1 090	8,7 min
Fe	^{54}Fe (n, p) ^{54}Mn	3,30 [c]	5	1	313 d
Fe	^{56}Fe (n, p) ^{56}Mn	6,10 [c] / 2,97 [a]	th	1 280	2,59 h
Fe	^{58}Fe (n, γ) ^{59}Fe	exoth [c]			44,5 d
Mn	^{55}Mn (n, 2n) ^{54}Mn	11,50 [c]	th	13 300	313 d
Mn	^{55}Mn (n, γ) ^{56}Mn	4,00 [c]			2,59 h
Ni	^{58}Ni (n, p) ^{58}Co, ^{58m}Co	2,80 [c]			71,3 d - 9,15 h
Ni	^{58}Ni (n, 2n) ^{57}Ni	12,30 [c]			36 h
Co	^{59}Co (n, γ) ^{60m}Co	exoth [a]	th	20 000	10,5 min
Co	^{59}Co (n, γ) ^{60}Co	exoth [a]	th	17 000	5,27 a
Cu	^{63}Cu (n, γ) ^{64}Cu	exoth [a]	th	4 500	12,7 h
Cu	^{63}Cu(n, 2n) ^{62}Cu	12,40 [c] / 10,03 [a]	11,5	10	9,76 min
Cu	^{65}Cu (n, γ) ^{66}Cu	exoth [a]	th	2 170	5,1 min
Cu	^{65}Cu (n, 2n) ^{64}Cu	11,20 [c] / 10,06 [a]	11	35	12,7 h
Zn	^{64}Zn (n, p) ^{64}Cu	exoth [a]			12,7 h
Rh	^{103}Rh (n, n') ^{103m}Rh	0,70 [c]			56,1 min
Ag	^{107}Ag (n, 2n) ^{106}Ag	10,70 [c]			24,1 min
Cd	^{113}Cd (n, γ) ^{114}Cd	exoth [a]	th	$2,06 \times 10^7$	
In	^{115}In (n, n') ^{115m}In	1,40 [c]			4,50 h
I	^{127}I (n, 2n) ^{126}I	11,50 [c]			12,8 d
Au	^{197}Au (n, γ) ^{198}Au	exoth [a]	th	$9,86 \times 10^4$	2,7 d
Th	^{232}Th(n, f) fission products	1,40 [c]			
Np	^{237}Np(n, f) fission products	0,70 [c]			
U	^{238}U(n, f) fission products	1,40 [c]			

[a] Extracted from “Schultz H., Voght H., Grundzüge des Praktischen Strahlenschutzses, 2., Auflage; Hanser, München-Wien”.

[b] Neutron produced by (p, n) reactions, with protons of 35 MeV.

[c] Extracted from “Neutron Contamination from Medical Electron Accelerators, National Council on Radiation Protection and Measurements”, Report No. 79, 1984.

σ: Reaction cross-section with neutrons of energy E_n

th: Thermal energy of neutrons

exoth: Exothermic reaction

Annex C
(informative)

Characteristics of the most common neutron shielding materials

This annex presents summarized tables listing the main characteristics of the most common neutron shielding materials. These materials are classified into organic and mineral materials.

Organic materials (see Table C.1).

- Polyethylenes
- Silicones
- Polyethylene/polypropylene resins
- Various organic materials

Mineral materials (see Table C.2).

- Concretes
- Mineral powders
- Water
- Mineral glasses
- Metallic alloys
- Hydrides
- Metal/resin compounds

The characteristics given in these tables concern the following information.

- Designation of material, with possibly the most common trade® name.
- Description of the material.
- Relative density.
- The main atomic contents, expressed in atoms/m^3.
- Percentage of principal additives, regarding the neutron and/or gamma absorption or attenuation.
- The main mechanical characteristics.
- Chemical resistance.
- Ease of decontamination.
- The maximal permissible temperature.
- The burning behaviour.
- Others characteristics, such as available forms of material, its workability, its fire load, the possible need for a casing, the resistance to radiation, etc.

NOTE 1 The information given in this annex concerns material that was commercially available at the time of writing this International Standard. For this reason, trade names ® are included. Inclusion of a specific material in this annex does not constitute an ISO approval or recommendation for its use. Similarly, the fact that a specific trade marked material is not included, shall not be interpreted as an ISO disapproval of its use. Manufacturers wishing to have materials included in the next revision of this International Standard should contact ISO.

NOTE 2 The information given in annex C is extracted from the manufacturers' technical files and is only given on their responsibility. End users who intend to use these materials shall verify for themselves, either by requesting the appropriate certificates proving the claimed performances or characteristics and delivered by independent test houses or by making their own approval tests before use.

NOTE 3 The nuclide contents that are underlined in Tables C.1 and C.2 correspond to an element playing a major role in the attenuation or absorption of neutrons (hydrogenous component, boron, cadmium) or in the attenuation of gamma radiation (lead, uranium, gadolinium).

Table C.1A — Main characteristics of organic materials (polyethylene family)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Pure polyethylene type ertalene ®	Pure high density polyethylene, consisting of a polymer (CH_2) chain	0,95	H: $\underline{8{,}18 \times 10^{-2}}$ C: $4{,}08 \times 10^{-2}$	No additives for pure material	Breaking stress (traction): 42 MPa (compression): not available	Moderate resistance to acids, oxidizing agents and solvents	Depends on the surface condition	80 °C	Burnable material	Available in slabs and easily workable. Very high fire load. Medium resistance to radiation. It can be charged with fire proofing elements and protected by a casing.
Borated polyethylene type 207 ®	Borated, self-extinguishing high density polyethylene	1,40	H: $\underline{6{,}61 \times 10^{-2}}$ B: $\underline{5{,}45 \times 10^{-4}}$ C: $2{,}42 \times 10^{-2}$ O: $1{,}92 \times 10^{-2}$	Boron (0,7 %)	Breaking stress (traction): 2,3 MPa (compression): 1,6 MPa	Like pure polyethylene	Good ease of decontamination	90 °C	Self extinguishable material	Available in slabs or blocks and easily workable. Very high fire load. Excellent fire resistance. Medium resistance to radiation. It can be protected by a casing.
Lead-loaded polyethylene Type 206 ®	Lead-loaded polyethylene for mixed γ and neutron radiation protection	2,92	H: $\underline{5{,}91 \times 10^{-2}}$ C: $2{,}92 \times 10^{-2}$ Pb: $\underline{6{,}49 \times 10^{-3}}$	Lead (76,5 %)	Breaking stress (traction): 0,6 MPa (compression): not available	Like pure polyethylene	Not very easy to decontaminate on non protected surface	85 °C	Burnable material	Available in slabs or blocks and easily workable. Very high fire-load. Medium resistance to radiation. It can be charged with fire proofing elements and protected by a casing.
Lead loaded polyethylene type 239 ®	Lead loaded polyethylene for mixed γ and neutron radiation protection	5,16	H: $\underline{2{,}58 \times 10^{-2}}$ C: $1{,}40 \times 10^{-2}$ O: $4{,}68 \times 10^{-3}$ Pb: $\underline{1{,}35 \times 10^{-2}}$	Lead (90,3 %)	Not yet available	Like pure polyethylene	Not very easy to decontaminate on non protected surface	85 °C	Burnable material	Available in flexible foils, covered by a PVC pocket. Very high fire-load. Medium resistance to radiation. Suitable for removable shielding. Easily workable.

Table C.1A — Main characteristics of organic materials (polyethylene family) *(continued)*

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Borated and lead-loaded polyethylene type 202 ®	Borated and lead-loaded polyethylene for mixed γ and neutron radiation protection	4,20	H: $\underline{2{,}58 \times 10^{-2}}$ C: $2{,}17 \times 10^{-2}$ O: $5{,}69 \times 10^{-3}$ Pb: $\underline{9{,}92 \times 10^{-3}}$ B: $\underline{2{,}43 \times 10^{-3}}$	Lead (81,33 %) Boron (1,04 %)	Breaking stress (traction): 0,5 MPa (compression): not available	Like pure polyethylene	Not very easy to decontaminate on non protected surface	85 °C	Burnable material	Available in blocks and easily workable. Very high fire load. Medium resistance to radiation. It can be charged with fire proofing elements and protected by a casing.
Hypalon with gadolinium oxide type LD 2069 ®	Chloro-sulfonated polyethylene, loaded with gadolinium oxide.	2,15	H: $\underline{4{,}90 \times 10^{-2}}$ C: $2{,}46 \times 10^{-2}$ Gd: $\underline{4{,}30 \times 10^{-3}}$ O: $6{,}70 \times 10^{-3}$	Gadolinium (52 %)	Not yet available	Good resistance to acids, oxidizing agents and solvents	Depends on the surface condition	90 °C	Burnable material	Only available in flexible foils. High fire-load. Medium resistance to radiation. It can be glued. Easily workable.
Borated silicone elastomer type 238 ®	Borated, self-extinguishing and high temperature resistant silicone elastomer	1,64	H: $\underline{2{,}72 \times 10^{-2}}$ B: $\underline{2{,}31 \times 10^{-2}}$ C: $1{,}65 \times 10^{-2}$ O: $1{,}49 \times 10^{-2}$	Boron (0,7 %)	Breaking stress (traction): not available (compression): 0,7 MPa	Moderate resistance to acids, oxidizing agents and solvents	Good ease of decontamination	205 °C	Self extinguishable material	Only available in flexible foils. High fire-load. Excellent fire resistance. Medium resistance to radiation. It can be glued and is easily workable.
Silicone elastomer type Sylgard ®	High temperature and fire resistant silicone elastomer	1,38	H: $\underline{3{,}55 \times 10^{-2}}$ C: $1{,}19 \times 10^{-2}$ O: $1{,}87 \times 10^{-2}$	No additives for pure material	Breaking stress (traction): 2,3 MPa (compression): not available	Low resistance to acids, oxidizing agents and solvents	Good ease of decontamination	150 °C	Non flammable material	Available in blocks or flexible foils; can also be cast on site. Easily workable by cutting or moulding. Can be loaded with lead, boron or polyethylene. High fire-load. Good fire resistance. Medium resistance to radiation.

Table C.1A — Main characteristics of organic materials (polyethylene family) (*continued*)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Borated silicone elastomer type Boraflex ®	Borated, high temperature and fire resistant silicone elastomer	1,70	H: $\underline{3,05 \times 10^{-2}}$ C: $1,61 \times 10^{-2}$ O: $1,40 \times 10^{-3}$ B: $\underline{2,98 \times 10^{-2}}$	Boron (31,5 %)	Breaking stress (traction): 1,4 MPa (compression): not available	Low resistance to acids, oxidizing agents and solvents	Good ease of decontamination	170 °C	Non burnable material	Available in flexible foils; can also be cast on site. Easily workable, by cutting or moulding. Good fire resistance. High fire-load. Medium resistance to radiation.
Borated or lead-loaded silicone elastomer type NS1 ®	Borated and/or lead-loaded, with high temperature resistant silicone elastomer	1,43	H: $\underline{3,60 \times 10^{-2}}$ C: $1,30 \times 10^{-2}$ O: $1,90 \times 10^{-2}$ B: $\underline{4,00 \times 10^{-2}}$	Boron (5,00 %) Lead can be added (1)	Breaking stress (traction): 3,5 MPa (compression): not available	Moderate resistance to acids, oxidizing agents and solvents	Good ease of decontamination	200 °C	Burnable material	Available in blocks or flexible foils; can also be cast on site. Easily workable by cutting or moulding. Boron or lead content is adjustable (1). High fire-load. Medium resistance to radiation.
Borated silicone elastomer type 237 ®	Borated, high-temperature and fire resistant foam of silicone elastomer	1,59	H: $\underline{4,52 \times 10^{-2}}$ C: $8,60 \times 10^{-3}$ O: $2,80 \times 10^{-2}$ B: $\underline{9,36 \times 10^{-4}}$	Boron (1,06 %)	Breaking stress (traction): 0,4 MPa (compression): 3,15 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	205 °C	Self extinguishable material	Available in blocks; can also be cast on site. Easily workable by cutting or filling casing with silicone powder Excellent fire resistance. High fire-load. Medium resistance to radiation.
Silicone elastomer type FS03 ®	High-temperature resistant foam of silicone elastomer	0,24	H: $\underline{8,67 \times 10^{-3}}$ C: $5,18 \times 10^{-3}$ O: $2,08 \times 10^{-3}$	No additives for pure material (1)	Breaking stress (traction): not available (compression): 0,23 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	150 °C	Burnable material	Available in blocks; can also be cast on site. Easily workable by cutting or moulding. Can be loaded with lead, boron or polyethylene (1). High fire-load. Medium resistance to radiation.

Table C.1B — Main characteristics of organic materials (polyethylene/polyethylene resins)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Neutronic compound, with polyethylene resin type 21 ®	Compound based on polyethylene, polyester resin, alumina and a borated material	1,25	H: $\underline{5,70 \times 10^{-2}}$ Al: $2,98 \times 10^{-3}$ O: $1,28 \times 10^{-2}$ B: $\underline{5,92 \times 10^{-4}}$	Boron (0,85 %)	Breaking stress (flexion): 6,3 MPa (compression): 28 MPa	Moderate resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	100 °C	Burning behaviour unknown	The material is available in blocks, which are not very easily workable. It can be protected by painting or using a casing. Medium resistance to radiation.
Neutronic compound, with polypropylene resin type 22 ®	Compound based on polypropylene, polyester resin, alumina and a borated material	1,20	H: $\underline{5,40 \times 10^{-2}}$ Al: $2,97 \times 10^{-3}$ O: $1,26 \times 10^{-2}$ B: $\underline{5,70 \times 10^{-4}}$	Boron (0,85 %)	Breaking stress (flexion): 6,3 MPa (compression): 20 MPa	Moderate resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	130 °C	Burnable material	The material is available in blocks, which are not very easily workable. It can be protected by painting or using a casing. Medium resistance to radiation.
Neutronic compound with polyethylene	Compound based on polyethylene with aluminium and boron	1,44	H: $\underline{6,01 \times 10^{-2}}$ Al: $5,61 \times 10^{-2}$ O: $1,71 \times 10^{-2}$ B: $\underline{1,71 \times 10^{-2}}$	Boron (2,0 %)	Breaking stress (traction): 11 MPa (compression): 28 MPa	Good resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	80 °C	Burnable material	Available in blocks and easily workable. It can be protected by painting or using a casing. Medium resistance to radiation.
Neutronic compound with polyamide resin	Compound based on polyamide with boron	1,36	H: $\underline{2,10 \times 10^{-2}}$ C: $2,70 \times 10^{-2}$ N: $3,00 \times 10^{-2}$ O: $9,00 \times 10^{-2}$ B: $\underline{2,20 \times 10^{-2}}$	Boron (3,0 %)	Breaking stress (traction): not available (compression): 20 MPa	Moderate to good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	180 °C	Burnable material	Manufactured by compressing. The material is available in blocks, which are easily workable. It can be protected by painting or using a casing. Medium resistance to radiation.

Table C.1C — Main characteristics of organic materials (transparent organic materials)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Poly-carbonate (Lexan ®, Makrolon ®)	Thermoplastic polymer	1,20	H: $3{,}98 \times 10^{-2}$ C: $4{,}55 \times 10^{-2}$ O: $8{,}54 \times 10^{-3}$	No additives for pure material	Breaking stress (traction): 70 MPa (compression): not available	Moderate to good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	120 °C	Burnable material	Only available in slabs. Easily workable. It can be brittle and easily scratched. Medium resistance to radiation. High fire-load.
Polymethyl methacrylate (Altuglas ®, Plexiglas ®)	Thermoplastic polymer	1,18	H: $5{,}77 \times 10^{-2}$ C: $3{,}55 \times 10^{-2}$ O: $1{,}42 \times 10^{-3}$	No additives for pure material	Breaking stress (traction): 69 MPa (compression): not available	Moderate to good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	100 °C	Burnable material	Available in slabs or blocks. It is easily workable and can be glued. It can be brittle and easily scratched. Low resistance to radiation. High fire-load.
Lead-loaded polymethyl methacrylate type Kiowaglas ®	Lead-loaded transparent thermoplastic polymer	1,60	H: $5{,}59 \times 10^{-2}$ C: $3{,}55 \times 10^{-2}$ O: $1{,}23 \times 10^{-2}$ Pb: $1{,}39 \times 10^{-3}$	Lead (30 %)	Breaking stress (traction): 49 MPa (flexion): 68 MPa	Moderate resistance to acids, oxidizing agents and solvents	Good ease of decontamination	80 °C	Burnable material	Available in slabs or blocks. It is easily workable and can be glued. It can be brittle and easily scratched. Low resistance to radiation. High fire-load.

Table C.1D — Main characteristics of organic materials (various organic materials)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Paraffin ®	Polymer chain (C_x-H_y)	0,90	H: $\underline{7,59 \times 10}^{-2}$ C: $3,88 \times 10^{-2}$	No additives for pure material	Not yet available	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	55 °C	Burnable material	The material can be cast on site by moulding or cutting. It needs a casing. Low resistance to radiation. Very high fire-load.
Asphalt	Compound with mixed hydrocarbides	1,00	H: $\underline{6,02 \times 10}^{-2}$ C: $4,27 \times 10^{-2}$	No additives for pure material	Not yet available	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	50 °C	Burnable material	The material can only be cast on site by moulding. It needs a casing. Low resistance to radiation. High fire-load.
Non borated PERMALI type EN ®	Compressed wood, containing phenolic resins	1,25	H: $\underline{4,89 \times 10}^{-2}$ C: $3,45 \times 10^{-2}$ O: $1,79 \times 10^{-2}$	No additives for pure material	Breaking stress (traction): 70 MPa (compression): not available	Moderate to low resistance to acids, oxidizing agents and solvents	Depends on the surface condition	90 °C	Burnable material	Available in blocks. Easily workable. It can be charged with fire proofing elements and protected by a casing in order to improve chemical resistance and ease of decontamination. Medium resistance to radiation. Medium fire-oad.

Table C.1D — Main characteristics of organic materials (various organic materials) (*continued*)

Designation of material	**Description**	**Rela-tive density**	**Main contents** in 10^{30} atoms/m^3	**Additives** % of mass	**Main mechanical characteristics**	**Chemical resistance**	**Ease of decon-tamination**	**Maximal allowable tempe-rature**	**Burning behaviour**	**Other characteristics**
Borated PERMALI type HB ®	Borated, self-extinguishing compressed wood, containing phenolic resins	1,35	H: $\underline{5{,}20 \times 10^{-2}}$ C: $2{,}91 \times 10^{-2}$ O: $2{,}34 \times 10^{-2}$ B: $\underline{1{,}29 \times 10^{-3}}$	Boron (2,5 %)	Breaking stress (traction): 70 MPa (compression): 150 MPa	Moderate to low resistance to acids, oxidizing agents and solvents	Depends on the surface condition.	60 °C	Self extinguish-able material	Available in blocks. Easily workable. It can be charged with boron and protected by a casing in order to improve chemical resistance and ease of decontamination. Strong material. Low fire load. Excellent fire resistance. Medium resistance to radiation.
Polyurethane resin with gadolinium	Polyurethane resin loaded with gadolinium oxide	1,30	H: $\underline{5{,}48 \times 10^{-2}}$ C: $3{,}37 \times 10^{-2}$ O: $1{,}27 \times 10^{-2}$	Gadoli-nium (15 %)	Breaking stress (traction): 55 Mpa (compression): 110 MPa	Moderate to good resistance to acids, oxidizing agents and solvents	Depends on the surface condition	120 °C	Burnable material	The material can only be used as coating. It is applied by painting. Medium resistance to radiation. High fire-load.

Table C.2A — Main characteristics of mineral materials (family of concretes)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Ordinary concrete	Mineral compound, based on cement, sand, water and other additives	2,30	H: $\underline{6{,}92 \times 10}^{-3}$ Si: $1{,}32 \times 10^{-2}$ Ca: $3{,}76 \times 10^{-3}$ O: $4{,}35 \times 10^{-2}$ Al: $1{,}96 \times 10^{-3}$	No additives for pure material	Breaking stress (traction): 2,3 MPa (compression): 30 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	80 °C	Non burnable material	Available in blocks. Can also be cast on site. Good mechanical resistance, but difficult to work. Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance. Inexpensive.
Colemanite concrete	Borated mineral compound, based on cement, sand, water and other additives (colema-nite)	1,88	H: $\underline{3{,}61 \times 10}^{-2}$ B: $\underline{9{,}18 \times 10}^{-3}$ Ca: $4{,}84 \times 10^{-3}$ O: $4{,}25 \times 10^{-2}$ Al: $2{,}04 \times 10^{-3}$	Boron (8,75 %)	Breaking stress (traction): 3,2 MPa (compression): 19 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	80 °C	Non burnable material	Available in blocks. Can also be cast on site. Good mechanical resistance, but difficult to work. Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance.
Magnetite concrete (heavy concrete)	Mineral compound, based on cement, sand, water and other additives rich in iron (magnetite)	3,33	H: $\underline{3{,}28 \times 10}^{-2}$ Fe: $1{,}62 \times 10^{-2}$ Ca: $1{,}86 \times 10^{-2}$ Al: $1{,}62 \times 10^{-3}$	No additives for pure material	Breaking stress (traction): 3,0 MPa (compression): 35 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	80 °C	Non burnable material	Available in blocks. Can also be cast on site. Good mechanical resistance, but difficult to work. Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance.
Limonite concrete	Mineral compound, based on cement, sand, water and other additives rich in iron (limonite)	2,47	H: $\underline{1{,}44 \times 10}^{-2}$ Fe: $4{,}33 \times 10^{-3}$ Ca: $5{,}40 \times 10^{-3}$ O: $4{,}40 \times 10^{-2}$ Si: $8{,}00 \times 10^{-3}$ C: $3{,}20 \times 10^{-3}$	No additives for pure material	Breaking stress (traction): 3,9 MPa (compression): 39 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	80 °C	Non burnable material	Available in blocks. Can also be cast on site. Good mechanical resistance, but difficult to work. Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance.

Table C.2A — Main characteristics of mineral materials (family of concretes) (*continued*)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Barytine sand	Barium sulfate pellets	4,2(the.) 2,5(eff.)	H: $\underline{2,02 \times 10^{-2}}$ Ba: $1,01 \times 10^{-2}$ O: $5,03 \times 10^{-2}$ S: $1,01 \times 10^{-2}$	No additives for pure material	Not relevant	Good resistance to acids, oxidizing agents and solvents	Not relevant	150 °C	Non burnable material	Available in pellets. Needs a casing. Suitable for removable shielding and for fire cutting material.
Barytine concrete	Mineral compound, based on cement, water, barytine sand, and barytine gravel	3,30	H: $\underline{1,46 \times 10^{-2}}$ Ba: $6,80 \times 10^{-3}$ O: $4,46 \times 10^{-2}$ Ca: $2,51 \times 10^{-3}$ S: $\underline{6,80 \times 10^{-3}}$	No additives for pure material	Breaking stress (traction): not available (compression): 29 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	80 °C	Non burnable material	Available in blocks. Can also be cast on site. Good mechanical resistance, but difficult to work. Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance.
Serpentine concrete	Mineral compound, based on cement, borated water, pebbles and other additives	2,30	H: $\underline{1,32 \times 10^{-2}}$ Si: $8,20 \times 10^{-2}$ Ca: $2,70 \times 10^{-3}$ O: $3,96 \times 10^{-2}$ Al: $2,70 \times 10^{-4}$ Mg: $2,7 \times 10^{-4}$ B: $\underline{3,36 \times 10^{-3}}$	Boron (3,0 %)	Breaking stress (traction): not available (compression): 23,4 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	450 °C	Non burnable material	Available in blocks. Can also be cast on site. Good mechanical resistance, but difficult to work. Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance. Expensive.

Table C.2B — Main characteristics of mineral materials (water)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Pure water	Demineralized water without any additive	1,00	H: $\underline{6{,}69 \times 10}^{-2}$ O: $\underline{3{,}34 \times 10}^{-2}$	No additives for pure material	Not relevant	Not relevant	Not relevant	100 °C	Depends on the casing	Needs in all cases a casing which is generally in stainless steel or in plastic. Inexpensive.
Borated water	Demineralized water containing boric acid	1,00	H: $\underline{6{,}32 \times 10}^{-2}$ B: $\underline{1{,}11 \times 10}^{-3}$ O: $3{,}29 \times 10^{-2}$	Boron (2,0 %)	Not relevant	Not relevant	Not relevant	100 °C	Depends on the casing	Needs in all cases a casing which is generally in stainless steel or in plastic. Boron content can be adjusted. Inexpensive.

Table C.2C — Main characteristics of mineral materials (mineral powders)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Mineral compound type 277 ®	Borated, refractory compound, containing mineral powders and different liquids	1,68	H: $\underline{3{,}41 \times 10^{-2}}$ Al: $8{,}96 \times 10^{-3}$ Ca: $2{,}23 \times 10^{-3}$ O: $3{,}70 \times 10^{-2}$ B: $\underline{1{,}45 \times 10^{-3}}$	Boron (1,56 %)	Breaking stress (traction): 0,7 MPa (compression): 10 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	150 °C	Non burnable material	Available in slabs or blocks. Can also be cast on site. Difficult to work (hard material). Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance.
Mineral compound type NS3 ®	Refractory compound, containing mineral powders and an acrylic resin	1,76	H: $\underline{5{,}14 \times 10^{-2}}$ Al: $7{,}03 \times 10^{-3}$ Ca: $1{,}49 \times 10^{-3}$ O: $3{,}78 \times 10^{-2}$ C: $8{,}26 \times 10^{-3}$	No additives for pure material (1)	Breaking stress (traction): not available (compression): 30,4 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	90 °C	Non burnable material	Available in blocks or rods. Can also be cast on site. Good mechanical resistance, but difficult to work. Can be protected by painting or using a casing in order to improve ease of decontamination and chemical resistance. Boron or lead content can be added and adjusted in the range of 2 % to 90 % (1).

Table C.2C — Main characteristics of mineral materials (mineral powders) *(continued)*

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Neutronic compound based on plaster with borated polyethylene type No. 9 ®	Mineral compound, containing plaster with polyethylene and different boron additives (colemanite)	1,03	H: $\underline{5{,}01 \times 10}^{-2}$ B: $\underline{6{,}47 \times 10}^{-4}$ Ca: $1{,}77 \times 10^{-3}$ O: $1{,}13 \times 10^{-2}$ C: $2{,}11 \times 10^{-2}$	Boron (1,13 %)	Breaking stress (traction): 1 MPa (compression): 4,5 MPa	Low resistance to water, acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	70 °C	Unknown	Available in blocks or rods. Can also be cast on site. Difficult to work. Because of brittle material and in order to improve mechanical and chemical resistance and ease of decontamination a casing is necessary. High fire-load.
Neutronic compound based on plaster with borated polypropylene type No. 10 ®	Mineral compound, containing plaster with polypropylene and different boron additives (colemanite)	0,99	H: $\underline{4{,}72 \times 10}^{-2}$ B: $\underline{6{,}20 \times 10}^{-4}$ Ca: $1{,}97 \times 10^{-3}$ O: $1{,}11 \times 10^{-2}$ C: $1{,}97 \times 10^{-2}$	Boron (1,13 %)	Breaking stress (traction): 1 MPa (compression): 4,5 MPa	Low resistance to water, acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	80 °C	Unknown	Available in blocks or rods. Can also be cast on site. Difficult to work. Because of brittle material and in order to improve mechanical and chemical resistance and ease of decontamination a casing is necessary. High fire-load.

Table C.2D — Main characteristics of mineral materials (hydrides)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Zirconium hydride with boron	Sintered zirconium hydride and boron	5,60	H: $\underline{7{,}23 \times 10^{-2}}$ Zr: $3{,}62 \times 10^{-2}$ B: $\underline{3{,}12 \times 10^{-2}}$	Boron (1,0 %)	Breaking stress (traction): 145 MPa (compression): 350 MPa	Low resistance to water, acids, oxidizing agents and solvents	Good ease of decontamination	550 °C	Burnable material	Available in slabs up to 100 × 200 × 50 mm which are packed in hermetically sealed containers. Very high fire load. The non protected material is easily workable. It needs a casing in order to improve chemical and fire resistance.
Titanium hydride	Compacted titanium hydride	3,80	H: $\underline{7{,}39 \times 10^{-2}}$ Ti: $4{,}62 \times 10^{-2}$ Fe: $0{,}78 \times 10^{-4}$ Al: $8{,}47 \times 10^{-4}$	No additives in pure material	Breaking stress (traction): not available (compression): 310 MPa	Low resistance to water, acids, oxidizing agents and solvents	Good ease of decontamination	400 °C	Burnable material	Available in slabs up to 100 × 200 × 50 mm which are packed in hermetically sealed containers. Very high fire load. Non protected material is easily workable. It needs a casing in order to improve ease of decontamination, and resistance to chemicals and fire.
Titanium hydride with boron	Compressed powder of titanium hydride with boron	3,30	H: $\underline{6{,}69 \times 10^{-2}}$ Ti: $3{,}87 \times 10^{-2}$ Fe: $4{,}32 \times 10^{-4}$ Mg: $1{,}63 \times 10^{-4}$ B: $\underline{5{,}52 \times 10^{-3}}$	Boron (3,0 %)	Breaking stress (traction): not available (compression): 310 MPa	Not yet available	Not very easy to decontaminate on non protected surface	250 °C	Burnable material	Available in slabs up to 100 × 200 × 50 mm which are packed in hermetically sealed containers. Very high fire load. Non protected material is easily workable. It needs a casing in order to improve ease of decontamination and resistance to chemicals and fire.

Table C.2D — Main characteristics of mineral materials (hydrides) (*continued*)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Lithium hydride	Compressed or cast lithium hydride	0,72	H: $\underline{5{,}46 \times 10^{-2}}$ Li: $5{,}46 \times 10^{-2}$	No additives in pure material	Breaking stress (flexion): 110 MPa (compression): 75 MPa	Non protected material highly reactive with water and humid air	Cannot be decontaminated	550 °C	Burnable material	Available only in hermetically sealed stainless steel containers. Very high fire-load. Highly reactive with water and humid air.
Diboric titanium	Compressed powder of diboric titanium	4,00	Ti: $3{,}58 \times 10^{-2}$ Fe: $4{,}32 \times 10^{-4}$ C: $1{,}40 \times 10^{-2}$ B: $\underline{7{,}16 \times 10^{-2}}$	Boron	Breaking stress (flexion): 130 MPa (compression): 350 MPa	Good resistance to water, acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	700 °C	Burnable material	Available in slabs up to 100 × 200 × 50 mm which are packed in hermetically sealed containers. Very high fire-load. Non protected material is easily workable. It needs a casing in order to improve ease of decontamination, and resistance to chemicals and fire.

Table C.2E — Main characteristics of mineral materials (lead loaded mineral glasses)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Lead-loaded glass type RS 323 G19 ®	Stabilized glass, with cerium oxide, loaded with lead oxide	3,26	Si: $1{,}43 \times 10^{-2}$ O: $3{,}83 \times 10^{-2}$ B: $\underline{1{,}69 \times 10^{-3}}$ Ce: $2{,}05 \times 10^{-4}$ Pb: $\underline{2{,}93 \times 10^{-3}}$	Lead (30,9 %)	Breaking stress (tensile strength): > 6 MPa (compression): > 2 GPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	250 °C	Non flammable material	Available in slabs or blocks. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. High resistance to radiation (browning and discharge resistant). Medium resistance to temperature. Excellent optical properties.
Lead-loaded glass type RS 360 ®	Transparent glass, loaded with lead oxide	3,60	Si: $1{,}65 \times 10^{-2}$ O: $3{,}99 \times 10^{-2}$ Na: $2{,}59 \times 10^{-3}$ K: $2{,}35 \times 10^{-3}$ Pb: $\underline{4{,}41 \times 10^{-3}}$	Lead (42,1 %)	Breaking stress (tensile strength): > 6 MPa (compression): > 2 GPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	250 °C	Non flammable material	Available in slabs or blocks. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. Medium resistance to radiation and to temperature. Excellent optical properties.
Lead-loaded glass type D 4.1 ®	Transparent glass, loaded with lead oxide	4,23	Si: $1{,}48 \times 10^{-2}$ O: $3{,}74 \times 10^{-2}$ Pb: $\underline{6{,}53 \times 10^{-3}}$	Lead (54,7 %)	Unknown	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	Unknown	Non flammable material	Available in slabs or blocks. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. Medium resistance to radiation and to temperature. Good optical properties.
Lead-loaded glass type RS 520 ®	Transparent glass, loaded with lead oxide	5,18	Si: $1{,}39 \times 10^{-2}$ O: $3{,}86 \times 10^{-2}$ Na: $5{,}03 \times 10^{-4}$ K: $6{,}62 \times 10^{-4}$ Pb: $\underline{1{,}00 \times 10^{-2}}$	Lead (66,5 %)	Breaking stress (tensile strength): > 6 MPa (compression): > 2 GPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	250 °C	Non flammable material	Available in slabs or blocks. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. Medium resistance to radiation and to temperature. Excellent optical properties.

Table C.2F — Main characteristics of mineral materials (ordinary mineral glasses and others)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Ordinary glass type 732 ®	Pyrex glass, made of borosilicate material	2,33	Si: $1{,}79 \times 10^{-2}$ O: $4{,}50 \times 10^{-2}$ B: $\underline{5{,}00 \times 10}^{-3}$ Al: $5{,}96 \times 10^{-4}$ Na: $1{,}37 \times 10^{-3}$	No additives for pure material	Breaking stress (traction): 15 MPa (compression): 360 MPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	400 °C	Non flammable material	Only available in slabs. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. Temperature resistant. Low resistance to radiation. Excellent optical properties.
Ordinary glass type RS 253 ®	Lead-free borosilicate glass	2,50	Si: $1{,}75 \times 10^{-2}$ O: $4{,}63 \times 10^{-2}$ B: $\underline{5{,}02 \times 10}^{-3}$ K: $2{,}33 \times 10^{-3}$ Na: $4{,}72 \times 10^{-3}$	No additives for pure material	Breaking stress (tensile strength): > 6 MPa (compression): > 2 GPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	400 °C	Non flammable material	Only available in slabs. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. Temperature resistant. Medium resistance to radiation. Excellent optical properties.
Stabilized glass type AS 25 18 ®	Stabilized glass, with cerium oxide	2,52	Si: $1{,}71 \times 10^{-2}$ O: $4{,}59 \times 10^{-2}$ B: $\underline{5{,}33 \times 10}^{-3}$ Ce: $1{,}59 \times 10^{-4}$ Na: $4{,}61 \times 10^{-3}$ K: $2{,}66 \times 10^{-3}$	No additives for pure material	Breaking stress (traction): not available (compression): 1 000 MPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	600 °C	Non flammable material	Available in slabs or blocks. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. High resistance to radiation (browning and discharge resistant) and to temperature. Excellent optical properties.

Table C.2F — Main characteristics of mineral materials (ordinary mineral glasses and others) (*continued*)

Designation of material	**Description**	**Relative density**	**Main contents** in 10^{30} atoms/m^3	**Additives** % of mass	**Main mechanical characteristics**	**Chemical resistance**	**Ease of decontamination**	**Maximal allowable temperature**	**Burning behaviour**	**Other characteristics**
Stabilized glass type RS 253 G18 ®	Stabilized glass, with cerium oxide	2,52	Si: $1{,}74 \times 10^{-2}$ O: $4{,}62 \times 10^{-2}$ B: $\underline{5{,}02 \times 10}^{-3}$ Ce: $1{,}59 \times 10^{-4}$ Na: $4{,}26 \times 10^{-3}$ K: $2{,}33 \times 10^{-3}$	No additives for pure material	Breaking stress (tensile strength): > 6 MPa (compression): > 2 GPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	400 °C	Non flammable material	Available in slabs or blocks. Easily workable. Low mechanical resistance. It can be brittle and easily scratched. High resistance to radiation (browning and discharge resistant) and to temperature. Excellent optical properties.
Permaglas type ME ®	Complex laminated glass/epoxy resin with gadolinium oxide	2,01	H: $\underline{1{,}72 \times 10}^{-2}$ O: $2{,}53 \times 10^{-3}$ Gd: $\underline{1{,}10 \times 10}^{-3}$ Si: $6{,}30 \times 10^{-3}$ C: $1{,}80 \times 10^{-2}$	Gadolinium (14,48 %)	Breaking stress (flexion): 370 MPa (compression): 325 MPa	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	155 °C	Burnable material	Only available in slabs form. Difficult to work because of brittle material. Bad mechanical resistance. Medium resistance to radiation. High fire-load.

Table C.2G — Main characteristics of mineral materials (pure metals)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Cadmium	Pure cadmium	8,65	^{110}Cd: $\underline{5,9 \times 10}^3$ ^{111}Cd: $\underline{6,0 \times 10}^3$ ^{112}Cd: $\underline{1,0 \times 10}^3$ ^{113}Cd: $\underline{5,6 \times 10}^3$ ^{114}Cd: $\underline{1,3 \times 10}^2$	No additives for pure material	Breaking stress (traction): 1 000 MPa (compression): non available	Cadmium is dissolved by any acid	Good ease of decontamination	320 °C	Non burnable material	Only available in thin foils. Because of its bad resistance to acids and of its toxicity, cadmium must be clad. Only used as anti criticality material, associated with a moderator.
Lead	Pure material (mild lead), or lead-antimony alloy (1)	11,00	Pb: $\underline{3,15 \times 10}^{-2}$ Sb: $2,25 \times 10^{-3}$	Lead (95,5 %) Antimony (4 %) (1)	Breaking stress (traction): 13 MPa (compression): 1,5 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	120 °C	Non burnable material	Available in blocks, foils, slabs, tubes or rods. Easily workable. In order to improve mechanical and chemical resistance and because of its toxicity, lead has to be clad. Antimony provides an increase in mechanical resistance (1).
Tungsten	Pure material or tungsten alloy, with very high fusion point	19,3	W: $\underline{6,32 \times 10}^{-2}$	Tungsten (99,98 %)	Breaking stress (traction): 1 000 MPa (compression): 1 150 MPa	Good resistance to acids, oxidizing agents and solvents	Unknown	3 000 °C	Non burnable material	Workable at high temperature on raw material by forging or laminating. To obtain finished pieces, final machining can be achieved at up to 200 °C. Welding is not very easy. Expensive.
Depleted uranium	High density depleted uranium	18,78	^{238}U: $\underline{4,74 \times 10}^2$ ^{235}U: $\underline{9,63 \times 10}^5$	No additives for pure material	Breaking stress (traction): 500 to 800 MPa (compression): 190 MPa	Uranium is dissolved by nitric acids and oxidized in air	Not very easy to decontaminate on non protected surface	500 °C	Oxidizing material	Available in slabs, rigid blocks or rods. Easily workable. Because of its ease of oxidizing, must be clad. Only used as anti criticality material, associated with a moderator.

Table C.2H — Main characteristics of mineral materials (metallic alloys)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Boron carbide	Sintered boron carbide	1,20 (powder) 2,52 (sintered)	B: $\underline{5{,}20 \times 10^{-2}}$ C: $1{,}29 \times 10^{-2}$	Boron (77,6 %)	Breaking stress (shear): 346 MPa (compression): not available	Good resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	2 350 °C	Non burnable material	Available in rigid blocks, slabs or rods. Easily workable. Because of brittleness, must be clad. Only used as anti criticality material, associated with a moderator.
Carbon steel	Iron-carbon alloy, with a carbon content of 0,05 % to 1,5 %	7,80	Fe: $8{,}27 \times 10^{-2}$ C: $7{,}82 \times 10^{-4}$ Mn: $8{,}55 \times 10^{-4}$ Si: $6{,}69 \times 10^{-4}$	No additives for pure material	Breaking stress (traction): 300 MPa to 600 MPa (compression): not available	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	1 500 °C	Non burnable material	Easily workable on raw material by moulding, forging, laminating, hot or cold drawing. To obtain finished pieces, final machining or welding can be achieved. The material should be protected by painting or using a casing in order to improve chemical resistance.
Stainless steel	Alloy steel with high chromium, nickel or chromium-nickel content	7,80	Fe: $6{,}03 \times 10^{-2}$ C: $0{,}78 \times 10^{-4}$ Cr: $1{,}63 \times 10^{-2}$ Ni: $0{,}79 \times 10^{-2}$	Nickel (10 %) Chromium (18 %) (1)	Breaking stress (traction): 450 MPa to 650 MPa (compression): not available	Good resistance to acids, oxidizing agents and solvents	Good ease of decontamination	1 400 °C	Non burnable material	Easily workable on raw material by moulding, forging, laminating, or hot or cold drawing. To obtain finished pieces, final machining or welding can be achieved. Present form corresponds to Z2 CN 18-10 (NF) or 304 L (AISI) (1).
Cast iron	Iron-carbon alloy, with a carbon content of 2,5 % to 4,5 %	7,2	Fe: $7{,}14 \times 10^{-2}$ C: $1{,}12 \times 10^{-2}$ Si: $3{,}47 \times 10^{-3}$ Mn: $5{,}92 \times 10^{-4}$ P: $9{,}10 \times 10^{-4}$	No additives for pure material	Breaking stress (traction): 100 to 400 MPa (compression): 600 to 900 MPa	Low resistance to acids, oxidizing agents and solvents	Not very easy to decontaminate on non protected surface	1 000 °C	Non burnable material	Easily workable on raw material by moulding. To obtain finished pieces, final machining can be achieved. Welding is not very easy. The material should be protected by painting or using a casing in order to improve chemical resistance.

Table C.2l — Main characteristics of mineral materials (metal/resin compounds)

Designation of material	Description	Relative density	Main contents in 10^{30} atoms/m^3	Additives % of mass	Main mechanical characteristics	Chemical resistance	Ease of decontamination	Maximal allowable temperature	Burning behaviour	Other characteristics
Heavy material, with iron filings	Polyester resin, with iron filings and borated acid	5,22	H: $\underline{1{,}54 \times 10^{-2}}$ O: $8{,}44 \times 10^{-3}$ B: $\underline{8{,}20 \times 10^{-3}}$ Fe: $5{,}01 \times 10^{-2}$ C: $1{,}62 \times 10^{-2}$	Boron (0,77 %)	Breaking stress (traction): 2,65 MPa (compression): 13,7 MPa	Unknown. Depends on the nature of the resin	Not very easy to decontaminate on non protected surface	200 °C	Unknown	Available in blocks. Can also be cast on site. Easily workable. Iron filings can be replaced by lead shot. Suitable for mixed γ and neutron radiation protection.
Heavy material, with iron and lead filings type MP1 ®	Heavy compound, with iron and lead filings and borated resin binding	4,50	H: $\underline{1{,}80 \times 10^{-2}}$ O: $5{,}49 \times 10^{-3}$ B: $\underline{5{,}66 \times 10^{-3}}$ Fe: $2{,}01 \times 10^{-2}$ C: $1{,}18 \times 10^{-2}$ Pb: $\underline{6{,}02 \times 10^{-3}}$	Boron (2,26 %) Lead (46,02 %)	Breaking stress (traction): 70 MPa (compression): 120 MPa	Unknown. Depends on the nature of the resin	Not very easy to decontaminate on non protected surface	125 °C	Burnable material	Available in blocks. Can also be cast on site. Easily workable. Suitable for mixed γ and neutron radiation protection. Lead, iron and boron content can be adjustable.
Heavy material, with iron or lead filings type MP2 ®	Heavy compound, with iron or lead filings and borated silicate binding	3,20	H: $\underline{1{,}27 \times 10^{-2}}$ O: $1{,}81 \times 10^{-2}$ B: $\underline{1{,}20 \times 10^{-3}}$ Fe: $2{,}55 \times 10^{-2}$ Si: $2{,}80 \times 10^{-3}$ C: $4{,}50 \times 10^{-3}$	Boron (0,66 %) Lead can be added (1)	Not yet available	Unknown. Depends on the nature of the resin	Not very easy to decontaminate on non protected surface	200 °C	Non flammable material	Available in blocks. Can also be cast on site. Easily workable. Suitable for mixed γ and neutron radiation protection, and for fire-cutting element. Iron, lead and boron content can be adjustable (1).

Annex D
(informative)

Available computer codes for determination of neutron radiation protection shielding

D.1 Introduction

D.1.1 General

For the design of neutron shielding, the neutron radiation protection calculations described in D.1.2 to D.1.4 are to be considered.

D.1.2 Source evaluation codes

Here the following feature prominently:

a) source software that calculates the production of neutrons from specific nuclear reactions;

b) decay software that calculates the natural evolution in time of a population of radionuclides;

c) depletion software that calculates the transformation in time of a population of radionuclides subject to a particular flux.

Categories b) and c) draw on nuclear data libraries that enable them to calculate, as a function of time among other things, the activity and/or source strengths of these populations.

D.1.3 Transport calculations

Here the aim is to evaluate a response in a point of the phase space, following the emission of particles in other points of this space.

Through evaluating particle flux of external exposure, among other things, these programs are used to design equipment and facilities, with regard to radiation protection.

D.1.4 Dose management software

Here the objective is to optimize the radiation protection of the workers as well as of the general public, with regard to the risk of exposure to ionizing radiation. The codes use databases that combine (measured or calculated) exposures with basic parameterized tasks or workplaces.

This software is used by the management to forecast evaluations of the external exposure of workers, and to calculate the dosimetric advantages gained by the installation when comparing different operating modes and considering different design options for the radiation protection shielding.

NOTE 1 End users may often use simplified software for calculating radiological impact. These programs (migration into geosphere, releases by facilities, individual and collective detriment) are based on methods that differ greatly from those of the radiation protection shielding calculations; therefore they are treated in detail in this International Standard.

NOTE 2 The information given in this annex concerns computer codes that were available at the time of writing this International Standard. The mention of these codes should not conclude that only those described can be used. Organizations in charge of the development of codes wishing have their codes included in the next revision of thIs International Standard should contact ISO.

D.2 Lists of the protection calculation codes, classified in alphabetical order and by type

D.2.1 Computer codes inventoried

See Table D.1.

Table D.1 — Codes inventoried

Name of code	Year of most recent update	Type [a] of code	Purpose [b]	Radiation [c] considered	Other codes to be associated upline or downline
ACORA	1993	G	D, P	γ, n	MERCURE, ANISN, INTERPOL
ALPHAN	1982	S	R, D, P	cp	No
ANISN	1994	T	D, P	γ, n	TAPEMA, PRESOU, MERCURE
CESAR	1993	S	R, D, P	n, e	No
DANTSYS 3.0		T	R, D, P		
DARWIN - PEPIN 2	1997	S	R, D, P	γ, n, e, α	APOLLO, PHADO, SN 1D, TRIPOLI, TWODANT, MERCURE
DOT/DORT	1994	T	R	γ, n	GIP, GRTUNEL
EFLUVE	1993	T	R	n	APOLLO
FAKIR	1993	S	R, D, P	activation	No
FLUKA	1996	T, G	R, D, P	γ, n, e, cp	No
HETC		T, G	R, D, P	γ, n	No
KAFKA	1993	S	P R, D, P	n, e	INTERPOL
MCBEND		T	R, D, P	γ, n	
MCNP	1997	T	R, D, P	γ, n, e	
MERCURE	1995	T	R, D, P	γ, n	No
MORSE		T	R, D, P	γ, n	
NARCISSE	1992	T	D, P	γ, n	No
ORIGEN	1991	S	R	γ, n, e, cp	No
SN 1D	1991	T	R, D, P	γ, n	No
TRABET A	1986	T	D, P	e	No
TRIPOLI	1997	T	R, D	γ, n	DARWIN, PEPIN 2, HETC
TWODANT	1990	T	D	γ, n	No

[a] S = Source; T = Transport; G = Dose Management.

[b] P = Basic design; D = Design; R = Research.

[c] γ = gamma; n = neutrons; e = electrons; p = protons; α = helium nuclei; cp = other charged particles.

D.2.2 Source evaluation codes

See Table D.2.

Table D.2 — Source evaluation codes

Name of code	Decay/Depletion				Reaction source	Calculated values [e]
	Type of nuclide [a]	Method [b]	$t_1 - t_2$ [c]	Results by [d]		
ALPHAN	AC	NI		Other	(α, n)	E_n
DARWIN - PEPIN 2	AC, PF, PA	NI, AN, DE	0,1 s to 10^9 a	I, E	All Types	C, S, P, A, E_n
FAKIR	AC, PF	Other	0,5 s to 10 a	Other		P, A
KAFKA	AC, PF, PA	NI	0,5 s to 10 a	I, E		A, Φ C, E_n, P
ORIGEN	AC, PF, PA	DE	0,1 s to 10^9 a	I, E	All types	C, S, P, A, E_n

[a] AC = Actinides; PF = Fission Products; PA = Activation Products.

[b] NI = Numerical; AN = Analytical; DE = Exponential Decomposition.

[c] Time interval for which the code is used.

[d] I = Isotope; E = Element.

[e] Φ is the fluence; A is the activity; S is the energy spectra; C is the composition (concentration, mass); P is the thermal decay heat (α, β, γ, total); E_n is the neutron production.

D.2.3 Transport calculation codes

Because of the considerable number of technical parameters used in this type of code, the presentation has been divided into three separate tables.

Table D.3 lists the calculated values and the theories and methods used for the calculation.

Table D.4 lists the sources and shielding used by each code.

Table D.5 lists the possibilities for analysing the contributions of certain parameters to the calculation results.

Table D.3 — Transport (values, theories and methods)

Name of code	Calculated values [a]	Theory [b] used	Method [c] of calculation	Dimensions of resolution	Time dependent
ALBATROS	D	α	δ	3	no
ANISN	Φ, D, S	B_{if}	δ, Y	1	no
DOT/DORT	Φ, J, S	B_{if}	δ, Y	2	no
EFLUVE	Φ, E_n	A_{tt}	δ	3	no
FLUKA	Φ, J, D, S	B_{if}	MC	3	yes
MCBEND	Φ, J, D, S	B_{if}	MC	3	yes
MCNP	Φ, J, D, S, E_n	B_{if}, B_i	MC	3	yes
MERCURE	Φ, D	R, A_{tt}	MC	3	no
MORSE	Φ, J, D, S, E_n	B_{if}, B_i	MC	3	yes
NARCISSE	D	α	δ	3	no
SN 1D	Φ, D, S	B_{if}	δ, Y	1	no
TRABETA	D	dE/dx	δ	1	no
TRIPOLI	Φ, J, D, S, E_n	B_{if}, B_i	MC	3	yes
TWODANT	Φ, J, D, S, E_n, A	B_{if}	δ, Y	1, 2	no

[a] Φ is the fluence; J is the current; S is the energy spectra; D is the absorbed dose, dose rate, dose equivalent rate, LET, kerma, kerma rate, energy deposited, damage; E_n is the neutron production; A is the activity.

[b] B_{if} = Boltzmann integral-differential; B_i = Boltzmann integral; A_{tt} = point kernel attenuation; α = reflection by albedo; dE/dx = stopping power.

[c] δ = numerical discrete; Y = spherical harmonics; MC = Monte Carlo.

Table D.4 — Transport (sources and shielding)

Name of code	Source term				Library	Shielding		
	Number	Geometry imposed	Transport		Number of materials or elements	Geometry imposed	Multi-layer (number)	Parameterization (thickness)
			Auto absorption	Shadow				
ALBATROS	single	imposed	no	no	materials (4)	imposed	yes	no
DOT/DORT	multiple	any	yes	yes	materials + elements	any	yes	yes
EFLUVE	multiple	any	yes	no	elements	imposed	no	no
FLUKA	multiple	any	yes	no	elements (25)	any	yes	no
MCBEND	multiple	any	yes	yes	elements (50)	any	yes	no
MERCURE	multiple	any	yes	yes	elements (50)	any	yes	yes
NARCISSE	multiple	imposed	no	no	elements (50)	imposed	no	no
SN 1D	multiple	any	yes	yes	elements (50)	any	yes	no
TRABETA	single	imposed	no	no	elements	imposed	no	no
TRIPOLI	multiple	any	yes	yes	elements (50)	any	yes	no
TWODANT	multiple	any	no	no	elements (50)	any	yes	no

Table D.5 — Transport (analysis of contributions)

Name of code	Contributions by		
	Source	Group	Shielding
ALBATROS	yes	yes	no
ANISN	no	yes	no
DOT/DORT	yes	yes	no
EFLUVE	yes	no	no
FLUKA	no	no	no
MERCURE	yes	yes	yes
NARCISSE	yes	no	no
SN 1D	no	yes	no
TRABETA	yes	no	no
TRIPOLI	no	yes	yes
TWODANT	no	yes	no

D.3 Dose management codes

The main characteristics of these codes are given in Tables D.6 and D.7.

Table D.6 lists the types of calculation and a description of the operational parameters studied.

Table D.7 lists the parameters analysed and the protection options used.

Table D.6 — Dose management (calculation, description of operations)

Name of code	Analysed doses [a]	Origin of flow rates [b]	Flowrate [c] $f(t)$	Description of operations [d]
ACORA	DI, DC	CC	no	LZ, TT, TE, CA
CERISE	DI	CC	SM, CD	LZ, TT, TE, SP, DE
DOSIANA	DC	SM	SM	LZ, TT, TE, SP

[a] DI = individual doses; DC = collective doses.

[b] SM = manual keying-in; CC = transport and attenuation code coupling.

[c] SM = manual keying-in; CD = heat decay calculation.

[d] SP = speciality; TT = total time period; TE = time of exposure; LZ = location/zoning; DE = wastes/effluents produced; CA = associated cost.

Table D.7 — Dose management (parameters analysed, protection options)

Name of code	Parameters analysed [a]		Protection options		
	Contribution	Sensitivity	Description of options [b]	Comparison criteria [c]	Chosen criteria
ACORA	OP, LZ	no	PS, LZ	DD, DI, DC, CA	yes
CERISE	OP, LZ, SP, TT	no	PS, DE, LZ, SP, TT	DD	no

[a] OP = operation; LZ = location/zoning; SP = speciality; TT = Total time period.

[b] PS = source protection; SP = speciality; TT = total time period; DE = wastes/effluents produced; LZ = location/zoning; SP = speciality.

[c] DD = dose equivalent rates; DI = individual dose; DC = collective doses; CA = associated cost.

Annex E
(informative)

Simplified shielding calculation method for neutrons produced by D-T reactions

E.1 Introduction

The purpose of this annex is to present a simplified method for the calculation of neutron shielding. This method is taken from NCRP Report N° 72. This approach could be followed for a mono-energetic source of neutrons provided attenuation curves were available for the desired material and the neutron energy.

E.2 Calculation

E.2.1 General

Simplified shielding calculations for D-T reactions are accomplished in three steps as described in E.2.2 to E.2.4.

1) calculation of the neutron fluence rate;

2) calculation of the ambient dose equivalent rate using the conversion coefficients $h^*(10)$;

3) calculation of the effect of the shielding upon the dose.

E.2.2 Neutron fluence rate, φ

The neutron fluence rate φ, for a neutron point source with an isotropic source strength B, in a vacuum, at a distance d from the source, is given by the following equation:

$$\varphi = \frac{B}{4 \times \pi \times d^2}$$

E.2.3 Converting neutron fluence to ambient dose equivalent rate, $\dot{H}^*(10)$

The ambient dose equivalent rate $\dot{H}^*(10)$, is given by:

$$\dot{H}^*(10) = \frac{B}{4 \times \pi \times d^2} \times h_{\Phi}^*(10)$$

The conversion coefficients $h_{\Phi}^*(10)$, according to the neutron energy, are given in Table E.1.

Table E.1 — Conversion coefficients $h_{\Phi}^*(10)$, from neutron fluence to ambient dose equivalent $\dot{H}^*(10)$ as a function of energy, in accordance with ICRP publication 74

Neutron energy MeV	**Conversion coefficient $h_{\Phi}^*(10)$** pSv·cm^2
0,001	7,9
0,01	10,5
0,05	41,1
0,1	88
0,5	322
1	416
2	420
3	412
5	405
7	405
10	440
12	480
14	520
20	600

E.2.4 Accounting for the effect of shielding

It is convenient to represent the effect of shielding by a transmission factor A in the above equation to give:

$$\dot{H}^*(10) = A \times \frac{B}{4 \times \pi \times d^2} \times h_{\Phi}^*(10)$$

The advanced calculating programs that are listed in annex D are required to determine the effects of absorption, scattering and energy degradation for neutrons in the calculation of the transmission factor. For large shielding, the transmission factor approaches the exponential form:

$$A \rightarrow B_\text{u} \times \mathrm{e}^{-(x/\lambda)}$$

where B_u is a constant called the build-up factor, x is the shielding thickness and λ is a constant called the relaxation length.

Table E.2 lists the approximate ambient dose equivalent transmission factors A, for 14 MeV neutrons, in ordinary concrete and water.

Table E.2 — Ambient dose equivalent transmission factors A, for neutrons of 14 MeV, in ordinary concrete and water [6)]

Transmission factor	Thickness cm	
	Ordinary concrete	Water
0,1	55	61
0,05	64	71
0,02	76	84
0,01	84	94
5×10^{-3}	94	104
2×10^{-3}	105	118
1×10^{-3}	114	128
5×10^{-4}	123	137
2×10^{-4}	135	150
1×10^{-4}	143	161
5×10^{-5}	152	171
2×10^{-5}	165	184
1×10^{-5}	174	195
5×10^{-6}	184	—
2×10^{-6}	197	—

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6) Radiation Protection Dosimetry, Vol. 58, No. 3, pp. 177-183 (1995).

Annex F
(informative)

Simplified neutron shielding calculation method

F.1 Introduction

The purpose of this annex is to describe the methodology for use of a simplified design method for a neutron protection shielding. Two methods are suitable:

- use of manual method (see clause F.2);
- use of abacuses (see clause F3).

F.2 Manual method

F.2.1 General

The following method is based on the point kernel method. It is applicable:

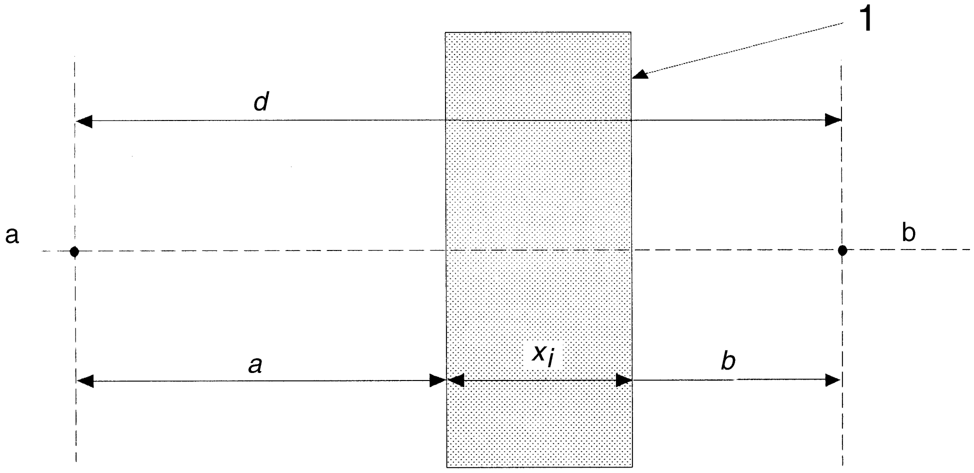
- for a point source;
- for a neutron emission spectrum identical to that used for the calculation of the coefficients μ and η (see F.2.3). This spectrum can be a fission spectrum, an (α, n) reaction spectrum or any other spectrum as defined in annex B;
- for homogeneous shielding, constituted from one single material.

NOTE According to the conversion coefficients used, this method permits the calculating of either the dose equivalent rates or the ambient dose equivalent rates. Nevertheless, in this annex, the results are expressed in terms of dose equivalent rates.

This method is entitled semi-manual because it needs either the use of an attenuation coefficient μ and a conversion coefficient η extracted from pre-established data bases or abacuses, or the determination, prior to the calculation, of these coefficients, according to the method presented in F.2.3, for the chosen neutron-protection shielding material (ordinary concrete, iron) and the considered neutron emission spectrum. The determination of the necessary coefficients for the manual calculation method is achieved using computer codes such as ANISN or SN 1D (see annex D).

F.2.2 Principle

See Figure F.1.



Key

1 Shielding

NOTE i = 1, 2 or 3.

[a] Source

[b] Calculation point

Figure F.1 — Principle of manual calculation method

A point source B, placed at a distance d from the calculation point gives a dose equivalent rate, $\dot{H}$, of following form through a protection shielding of thickness x:

$$\dot{H} = \frac{B \times \mathrm{e}^{-\mu x}}{4 \times \pi \times d^2} \times \eta$$

where

B is the neutron source strength in neutrons per second;

μ is the linear attenuation coefficient expressed as per metre;

x is the thickness of shielding material crossed in metres;

d is the distance between the point source and the calculation point in metres;

η is the coefficient taking into account the neutron fluence-to-dose equivalent conversion coefficient and the contribution of the diffused neutrons through the crossed shielding material in sieverts per hour per neutron per metre squared per second.

F.2.3 Calculation of μ and η

The coefficients μ and η are determined using a neutron transport code, for a specified neutron spectrum emission and for a given shielding material. The calculation is realized in two steps:

- calculation of average μ, in the range of the chosen thicknesses of the neutron shielding material;
- calculation of average η, from the previous calculation of the average μ value.

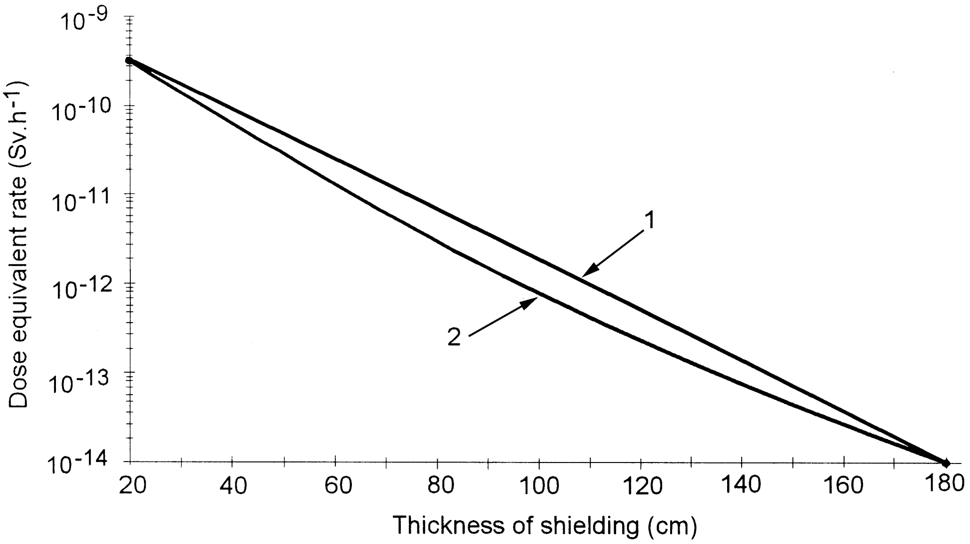
The calculation of the average attenuation is determined after achieving the calculation of three dose equivalent rates $\dot{H}_1$, $\dot{H}_2$, $\dot{H}_3$, for three thicknesses of shielding material x_1, x_2 and x_3.

The distance d between the source and the calculation point is constant. According to the indications given in Figure F.1, this distance d is equal to the sum of the distances $(a + x_i + b)$; it is higher than x_3 in any case.

The distances x_1, x_2 and x_3 are chosen (see Figure F.2) in such a way that:

— they are greater than the distances found in the typical application;

— the variations of $\dot{H}_1$, $\dot{H}_2$, $\dot{H}_3$, according to the thicknesses of the protection material are as near as possible an exponential function.



Key

1 Real curve, obtained using ANISN code

2 Theoretical exponential curve, determined from the previous real curve, which permits the calculation of the coefficients taken into account in the semi-manual calculation method

Figure F.2 — Relative variation of the dose equivalent rates, as a function of the thicknesses of the shielding material

The average μ value is calculated as follows:

$$\bar{\mu} = \frac{\mu_{12} + \mu_{23} + \mu_{13}}{3}$$

where

$$\mu_{12} = \frac{1}{x_2 - x_1} \times \ln\left(\frac{\dot{H}_1}{\dot{H}_2}\right)$$

$$\mu_{23} = \frac{1}{x_3 - x_2} \times \ln\left(\frac{\dot{H}_2}{\dot{H}_3}\right)$$

$$\mu_{13} = \frac{1}{x_3 - x_1} \times \ln\left(\frac{\dot{H}_1}{\dot{H}_3}\right)$$

The average η value is calculated as follows:

$$\bar{\eta} = \frac{\eta_1 + \eta_2 + \eta_3}{3}$$

where

$$\eta_1 = \frac{4\pi d^2 \times \dot{H}_1}{B} \times e^{\bar{\mu} x_1}$$

$$\eta_2 = \frac{4\pi d^2 \times \dot{H}_2}{B} \times e^{\bar{\mu} x_2}$$

$$\eta_3 = \frac{4\pi d^2 \times \dot{H}_3}{B} \times e^{\bar{\mu} x_3}$$

Table F.1 gives an example of calculation of the coefficients μ and η for several types of shielding material. These coefficients are determined:

- using SN 1D code;
- for a fission neutron source (^{235}U);
- with a distance d = 2 m;
- using neutron fluence-to-dose equivalent conversion coefficients issued from ANS 6.1.1 [1997].

Table F.1 — Determination of the coefficients μ and η

Shielding material	x_1, x_2, x_3 m	μ m^{-1}	η Sv·h^{-1}, $\text{n·m}^{-2}\text{·s}^{-1}$
Iron	0,1, 0,2, 0,3	4,764	$1{,}47 \times 10^{-10}$
PMMA	0,1, 0,2, 0,3	12,05	$1{,}01 \times 10^{-10}$
Ordinary concrete	0,5, 1,00, 1,50	33,39	$1{,}47 \times 10^{-10}$

NOTE The coefficients μ and η take into account the capture of the gamma radiation due to the (n, γ) reaction in the presence of the remaining neutrons in the shielding material.

F.2.4 Field of application

This method applies to any kind of shielding material (polyethylene, organic glasses, mineral powders, plasters, concrete, metallic alloys), on condition that pre-established data bases of η and μ exist (or that these coefficients can be determined during the calculation), and for any shielding thickness.

It can nevertheless only be used:

a) for the predetermination of neutron shielding or for the comparison of relative efficiency of different neutron shielding;

b) for homogeneous shielding, constituted by one single material;

c) for sources assimilated to point sources;

d) when the shielding thicknesses are comprised in the range of the thicknesses chosen for determining the μ and η coefficients;

e) for neutron spectrum emissions similar to those chosen for determining μ and η coefficients.

NOTE A point source in general gives higher dose equivalent rates than those from any other kind of geometry of sources, so this semi-manual method gives mostly excessively results.

F.2.5 Qualification of the method

The validation of the present method was done by the comparison of SN 1D code results and the results obtained with the manual method under the following conditions:

- point source of 1,00 × 10^{12} neutrons·s^{-1};
- fission spectrum of ^{235}U;
- material of shielding protection: ordinary concrete;
- distance between the source and the calculation point: 2 m;
- taking into account the gamma produced by the (n, γ) reaction;
- neutron fluence to dose equivalent conversion coefficients issued by ANS 6.1.1 [1997].

Table F.2 gives the results of this comparative study.

Table F.2 — Comparison between the results of SN 1D code and the results of the manual calculation

	x_1 = 50 cm, x_3 = 150 cm		x_1 = 20 cm, x_3 = 180 cm	
Thickness of concrete m	**SN 1D** µSv·h^{-1}	**Manual method** µSv·h^{-1}	**SN 1D** µSv·h^{-1}	**Manual method** µSv·h^{-1}
0,5	444	432	444	451
0,9	28,6	29,9	28,6	34,0

F.3 Abacuses

F.3.1 Principle

The abacuses are a set of graphs which gives the dose equivalent rates for different thicknesses of shielding system in a simplified situation, drawn up with SN 1D code.

The simplified situations are the following:

- Source geometry: point source
  linear infinite source
  massive semi-infinite medium
- Nature of source: fast neutron
  (α, n) reaction with ^{18}O
- Source medium (for semi-infinite medium only): uranyl nitrate of a concentration of 256 g·l^{-1}

- Shielding medium:

ordinary concrete	(d = 2,30)	⇒	30 cm	to	160 cm
heavy concrete	(d = 3,30)	⇒	30 cm	to	120 cm
cast iron	(d = 7,20)	⇒	10 cm	to	50 cm
PMMA	(d = 1,19)	⇒	0,2 cm	to	20 cm
water	(d = 1,00)	⇒	10 cm	to	400 cm
stainless steel	(d = 7,85)	⇒	0,2 cm	to	50 cm
lead glass	(d = 4,23)	⇒	1 cm	to	50 cm
borated polyethylene plaster No. 9	(d = 1,03)	⇒	5 cm	to	30 cm
borated polyethylene plaster No. 10	(d = 0,99)	⇒	5 cm	to	30 cm
polyethylene resin RP21	(d = 1,25)	⇒	5 cm	to	30 cm
polyethylene resin RP22	(d = 1,20)	⇒	5 cm	to	30 cm

- Shielding geometry: plane

- Distance between the calculation point and the source: 200 cm

- Neutron emission (B):

1 neutron$\cdot$s^{-1}	if point source
1 neutron$\cdot$s$^{-1}\cdot$cm^{-1}	if linear infinite source
1 neutron$\cdot$s$^{-1}\cdot$cm^{-3}	if massive semi-infinite medium

Figures F.3, F.4. and F.5 illustrate examples of such abacuses.

F.3.2 Method used

The abacuses are used as follows. Each abacus gives, for a thickness of shielding material and a given source, four terms that contribute to the dose equivalent rate:

- the contribution of spontaneous fission neutrons: $\dot{H}_{\mathrm{n}}$ (fs);
- the contribution of secondary gamma issuing from the spontaneous fission neutrons: $\dot{H}_{\gamma\mathrm{s}}$ (fs);
- the contribution of reaction neutrons (α, n): $\dot{H}_{\mathrm{n}}$ (α, n);
- the contribution of secondary gamma issuing from reaction neutrons (α, n): $\dot{H}_{\gamma\mathrm{s}}$ (α, n).

The four contributions should be added after having been multiplied by the source strength of the neutron source. However the greatest contribution of neutrons and secondary gamma could be used.

The dose equivalent rate due to the neutrons is then as follows:

$$\dot{H} = S \times \left(\dot{H}_{\mathrm{n}} + \dot{H}_{\gamma\mathrm{s}} \right)$$

where

S is the source strength of the neutron source (in n$\cdot$s^{-1} for a point source; n$\cdot$s$^{-1}\cdot$cm^{-1} for a linear infinite source and n$\cdot$s$^{-1}\cdot$cm^{-3} for a massive semi-infinite medium);

$\dot{H}_{\mathrm{n}} = \dot{H}_{\mathrm{n}}(\mathrm{fs}) + \dot{H}_{\mathrm{n}}(\alpha, \mathrm{n})$, or the greatest value between $\dot{H}_{\mathrm{n}}$(fs) and $\dot{H}_{\mathrm{n}}$(α, n);

$\dot{H}_{\gamma\mathrm{s}} = \dot{H}_{\gamma\mathrm{s}}(\mathrm{fs}) + \dot{H}_{\gamma\mathrm{s}}(\alpha, \mathrm{n})$, or the greatest value between $\dot{H}_{\gamma\mathrm{s}}$(fs) and $\dot{H}_{\gamma\mathrm{s}}$($\alpha$, n).

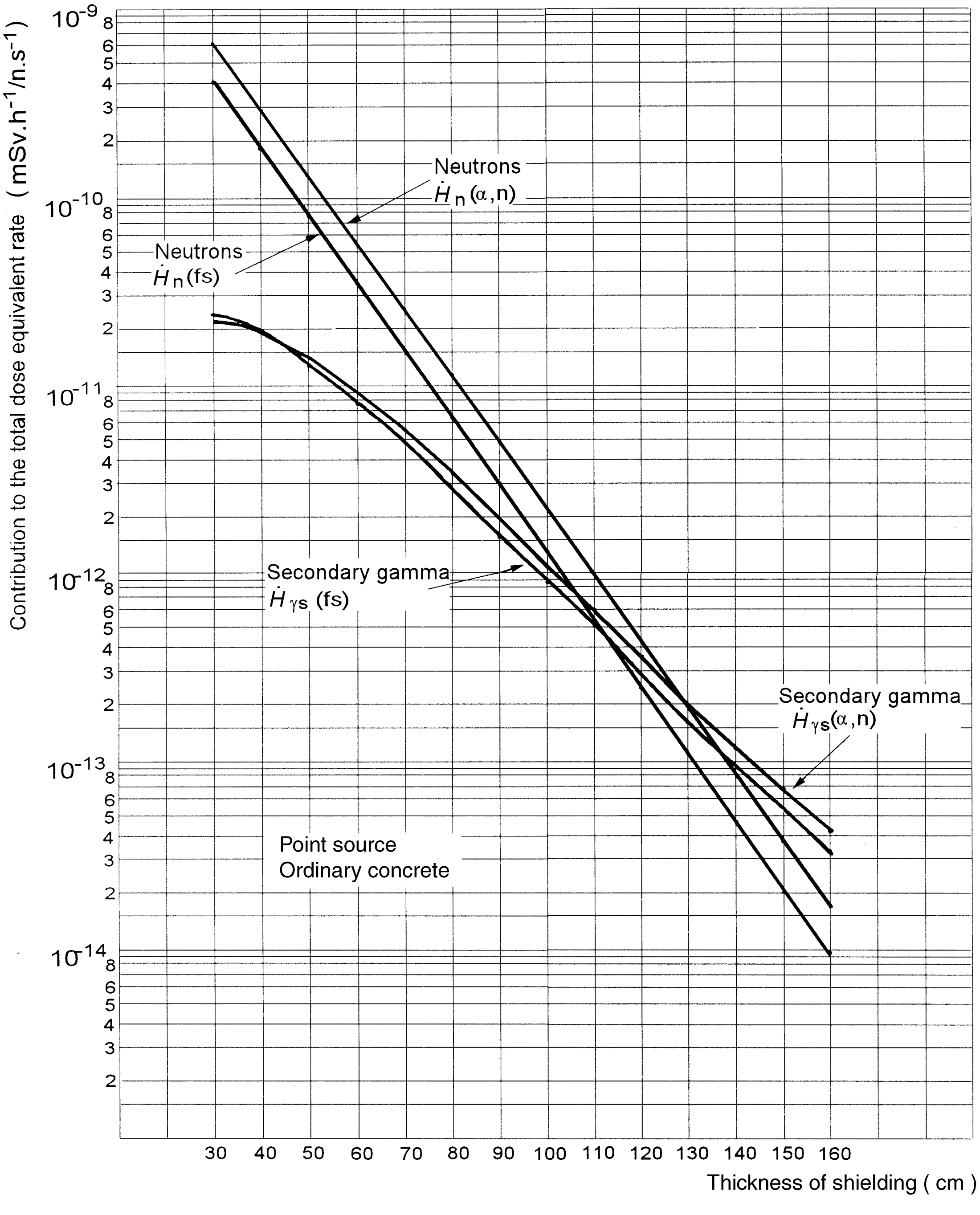


Figure F.3 — Abacus giving the respective contributions to the total dose equivalent rate, according to the thickness of the shielding (ordinary concrete) for a point source

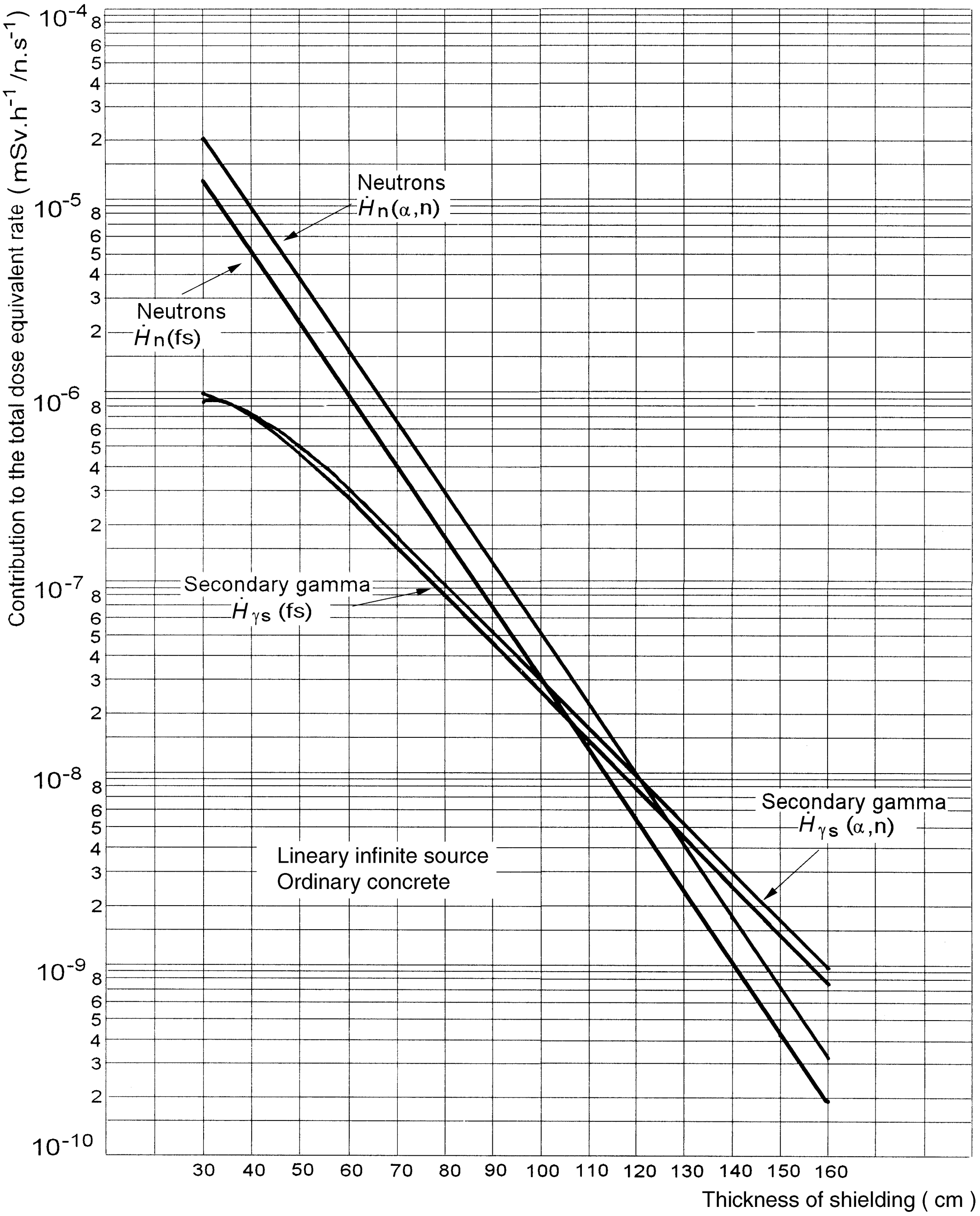


Figure F.4 — Abacus giving the respective contributions to the total dose equivalent rate, according to the thickness of the shielding (ordinary concrete) for a linear infinite source

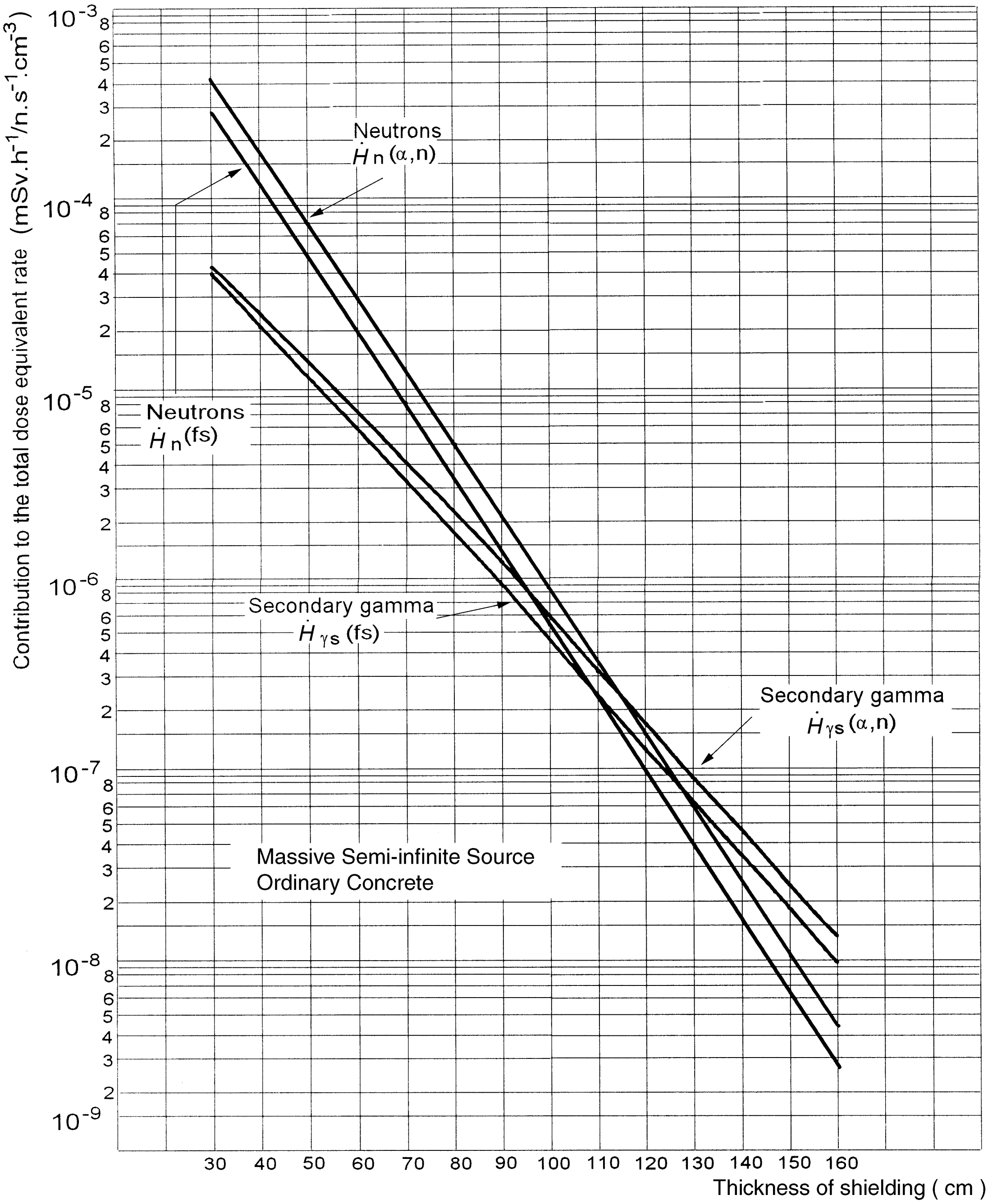


Figure F.5 — Abacus giving the respective contributions to the total dose equivalent rate, according to the thickness of the shielding (ordinary concrete) for a massive semi-infinite source

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