

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

ICS 13.280

National foreword

This British Standard reproduces verbatim ISO 14152:2001 and implements it as the UK national standard.

The UK participation in its preparation was entrusted to Technical Committee NCE/2, Health physics instrumentation, which has the responsibility to:

- aid enquirers to understand the text;
- present to the responsible international/European committee any enquiries on the interpretation, or proposals for change, and keep the UK interests informed;
- monitor related international and European developments and promulgate them in the UK.

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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 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

1) International Atomic Energy Agency.

3 Terms and definitions

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

3.1 Quantities²⁾ 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 ($\text{J}\cdot\text{kg}^{-1}$), 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 ($\text{J}\cdot\text{kg}^{-1}$), with the special name sievert (Sv).

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 ($\text{J}\cdot\text{kg}^{-1}$), 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 ($\text{J}\cdot\text{kg}^{-1}$), 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 $dH^*(d)$ per dt :

$$\dot{H}^*(d) = \frac{dH^*(d)}{dt}$$

where $dH^*(d)$ 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 ($\text{J}\cdot\text{kg}^{-1}\cdot\text{s}^{-1}$), with the special name sievert per second ($\text{Sv}\cdot\text{s}^{-1}$). A frequently used unit is sievert per hour ($\text{Sv}\cdot\text{h}^{-1}$).

3.2 Other definitions

3.2.1

neutron fluence

Φ

quotient of dN by da :

$$\Phi = \frac{dN}{da}$$

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 $d\phi$ by dt :

$$\phi = \frac{d\phi}{dt} = \frac{d^2N}{da \times dt}$$

where $d\phi$ 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 $m^{-2} \cdot s^{-1}$.**3.2.3****spectral neutron fluence
energy distribution of the neutron fluence** Φ_E quotient of $d\phi$ by dE :

$$\Phi_E = \frac{d\phi}{dE}$$

where $d\phi$ is the increment of neutron fluence in the energy interval between E and $E + dE$

[ICRU 51:1993, ISO 8529-1]

NOTE The SI unit of the spectral neutron fluence is $m^{-2} \cdot J^{-1}$; a frequently used unit is $cm^{-2} \cdot eV^{-1}$.**3.2.4****spectral neutron fluence rate
spectral neutron flux density** ϕ_E quotient of $d\phi_E$ by dt :

$$\phi_E = \frac{d\phi_E}{dt} = \frac{d^2\phi}{dE \times dt}$$

where $d\phi_E$ 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 $m^{-2} \cdot s^{-1} \cdot J^{-1}$; a frequently used unit is $cm^{-2} \cdot s^{-1} \cdot eV^{-1}$.**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 $\text{Sv}\cdot\text{m}^2$.

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 dN^* by dt :

$$B = \frac{dN^*}{dt}$$

where dN^* is the expectation value of the number of neutrons emitted by the source in the time interval dt

[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 :

$$B_E = \frac{dB}{dE}$$

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 $s^{-1} \cdot J^{-1}$; a frequently used unit is $cm^{-2} \cdot s^{-1} \cdot eV^{-1}$.

3.2.11 neutron flux

$\dot{N}$
quotient of dN by dt :

$$\dot{N} = \frac{dN}{dt}$$

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 :

$$K = \frac{dE_{tr}}{dm}$$

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 ($J \cdot kg^{-1}$), 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.

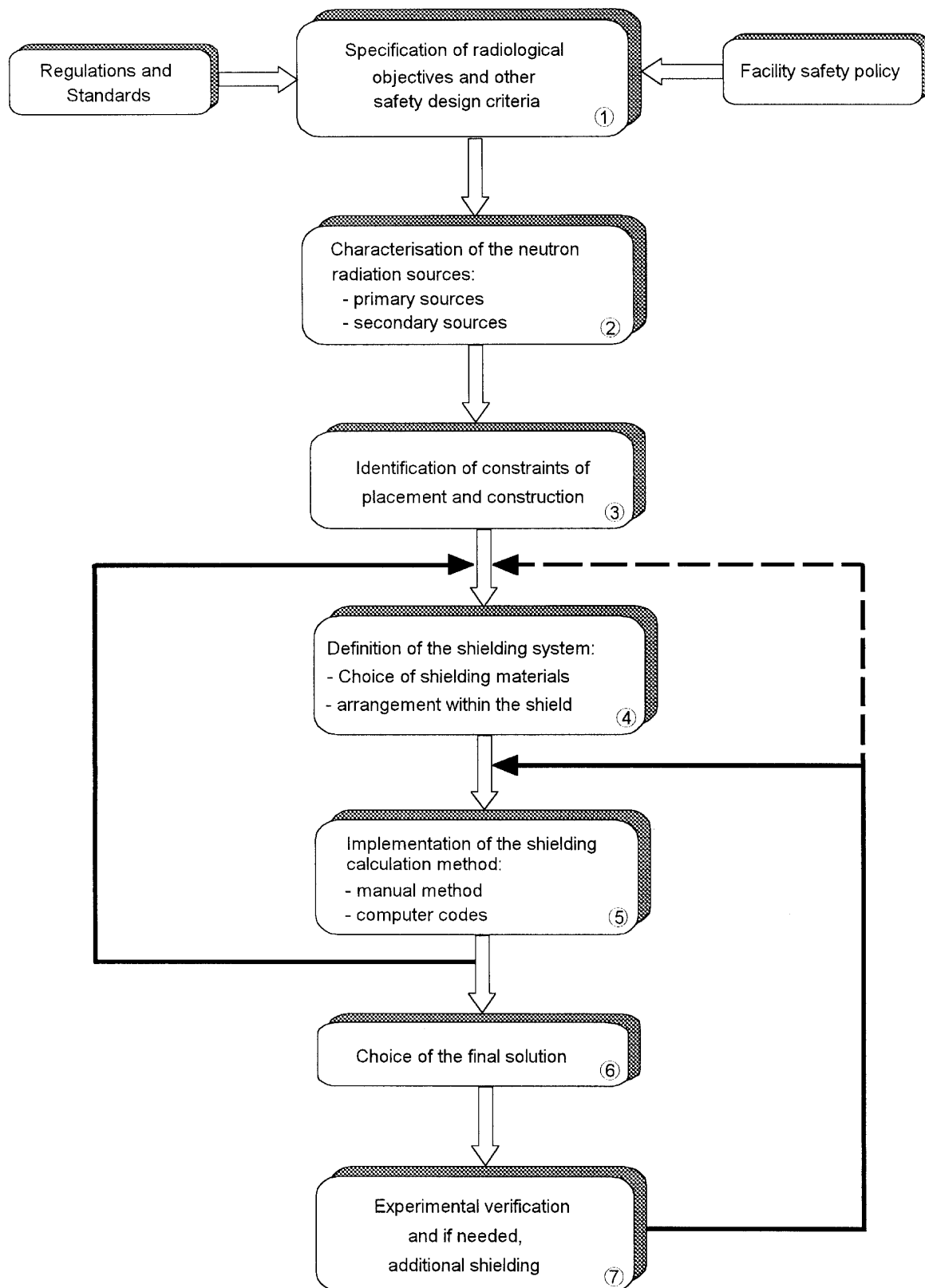


Figure 1 — General architecture of the methodology used to establish an operational neutron radiation protection shielding system

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 $400 \mu\text{Sv}\cdot\text{week}^{-1}$,
- an average of $10 \mu\text{Sv}\cdot\text{h}^{-1}$, for a 40-hour working week (or $2\,000 \text{ h}\cdot\text{a}^{-1}$).

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, supervised and restricted areas.
- b) Control of personnel access into the controlled, supervised 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, supervised 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 clad 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