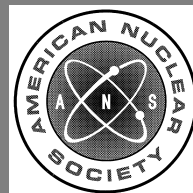


American Nuclear Society

**nuclear analysis and design
of concrete radiation shielding
for nuclear power plants**

an American National Standard



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**American National Standard
Nuclear Analysis and Design of Concrete
Radiation Shielding for Nuclear Power Plants**

Secretariat
American Nuclear Society

Prepared by the
**American Nuclear Society
Standards Committee
Working Group ANS-6.4**

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Foreword

(This Foreword is not a part of American National Standard “Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants,” ANSI/ANS-6.4-2006.)

The need for this standard was identified in mid-1972 by D. K. Trubey, then chairman of Subcommittee ANS-6, Radiation Protection and Shielding. The then-existing standard, ANSI N101.6-1972, “Concrete Radiation Shields,” provided excellent guidance on the construction of concrete radiation shielding structures but contained almost no information on shielding effectiveness or analysis. This standard was first issued as ANSI/ANS-6.4-1977 (N403).

After ANSI/ANS-6.4-1977 was issued, two significant events occurred that led to the decision to revise the standard: ANSI N101.6-1972 was withdrawn by ANSI, and the American Concrete Institute (ACI) issued its standard ACI 349-80, “Code Requirements for Nuclear Safety Related Concrete Structures,” as well as the Commentary ACI 349R-80, which provided updated requirements with regard to the construction aspects of concrete shielding structures. The withdrawal of ANSI N101.6-1972; the guidance provided by ACI 349-80; and advances in the evolution of shielding methods, data, and applications led to the revision, ANSI/ANS-6.4-1985.

Since that revision effort, advances in buildup factors prompted the revision ANSI/ANS-6.4-1997. Other advances, particularly with respect to transmission and reflection of gamma rays and neutrons by concrete slabs, prompted the current revision, ANSI/ANS-6.4-2006.

This revised standard is meant to be a “guide to good practice” in the area of concrete shielding analysis and design. Recommendations are given where possible, but more often the choice of analytical methods must be left to the discretion of the shielding engineer as appropriate to the particular job, whether it be a conceptual design or final construction drawing.

This standard was revised by Working Group ANS-6.4 of the American Nuclear Society, which had the following members at the time it prepared and approved this standard:

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Contents	Section	Page
	1	Scope
	1	1
	2	Requirements and recommendations
	2.1	Conformance
	2.2	Requirements
	2.3	Recommendations
	1	1
	2	2
	3	Standards of documentation
	3.1	Shield design approach
	3.2	Shield design description
	3.3	Methods of analysis
	3.4	Description of analyses
	3.5	References
	3.6	Summary results
	4	4
	4	Terms and definitions
	4	4
	5	Characterization of concrete
	5.1	Introduction
	5.2	Concrete placement
	5.3	Water content
	5.4	Heating effects
	5.5	Reinforcing steel
	5.6	Aggregates
	6	6
	7	7
	8	8
	8	8
	8	8
	6	Calculation methods
	6.1	Introduction
	6.2	Point-kernel methods
	6.3	Discrete ordinates method
	6.4	Monte Carlo method
	6.5	Other methods and summary
	10	10
	10	10
	10	10
	13	13
	16	16
	18	18
	7	Concrete shielding data
	7.1	Introduction
	7.2	Gamma-ray attenuation coefficients
	7.3	Gamma-ray buildup factors
	7.4	Secondary-gamma-ray production
	7.5	Neutron cross sections
	7.6	Neutron attenuation curves
	18	18
	18	18
	19	19
	19	19
	20	20
	20	20
	21	21
	8	Applications
	8.1	Radiation effects
	8.2	Minimum water content
	8.3	Bulk shielding
	8.4	Radiation streaming through penetrations
	8.5	Reflection
	21	21
	21	21
	22	22
	23	23
	25	25
	26	26
	9	References
	27	27
	Bibliography	32
	32	32
	Tables	
	Table 1	Typical concrete properties
	Table 2	Partial densities of concrete constituents
	3	3
	6	6

Figures

Figure 1	Example of oblique penetration short circuit paths	12
Figure 2	P_1 convergence of the fast neutron dose rate in water from a point fission source	15
Figure 3	Comparison of neutron and gamma-ray dose rate through concrete using transport and diffusion theory	17
Figure 4	Example of excess transmission through a void	25

Appendices

Appendix A	List of codes	33
Appendix B	Shielding data	36
Table B.1	Mass attenuation coefficients, excluding coherent scattering, for concretes described in Table 1 of the standard	37
Table B.2	Mass attenuation coefficients, excluding coherent scattering, for elements comprising ordinary concrete	38
Table B.3	Mass attenuation coefficients, excluding coherent scattering, for elements that may be found in other concrete compositions	39
Table B.4	Mass energy transfer coefficients for concretes described in Table 1 of the standard	40
Table B.5	Mass energy absorption coefficients for concretes described in Table 1 of the standard	41
Table B.6	Taylor coefficients for approximating concrete buildup factors	42
Table B.7	Geometric progression buildup factor coefficients for ordinary concrete (air kerma, 40 mfp)	43
Table B.8	Geometric progression buildup factor coefficients for ordinary concrete (concrete kerma, 40 mfp)	44
Table B.9	Gamma-ray spectra from thermal neutron capture in concrete	45
Table B.10	Neutron constants for concrete	45
Figure B.1	Air kerma buildup factors in concrete, moments method calculations	46
Figure B.2	Concrete kerma buildup factors in concrete, moments method calculations	47
Figure B.3	Air kerma buildup factors in concrete, geometric progression form (energy: 0.03 to 0.2 MeV)	48
Figure B.4	Air kerma buildup factors in concrete, geometric progression form (energy: 0.3 to 2.0 MeV)	49
Figure B.5	Air kerma buildup factors in concrete, geometric progression form (energy: 3.0 to 15.0 MeV)	50
Figure B.6	Concrete kerma buildup factors in concrete, geometric progression form (energy: 0.03 to 0.2 MeV)	51
Figure B.7	Concrete kerma buildup factors in concrete, geometric progression form (energy: 0.3 to 2.0 MeV)	52
Figure B.8	Concrete kerma buildup factors in concrete, geometric progression form (energy: 3.0 to 15.0 MeV)	53
Figure B.9	Comparison of adjoint discrete ordinates and moments method calculations of neutron dose equivalent (rem) from a plane isotropic fission source as a function of concrete thickness, source normalized to one neutron in a forward direction	54

Appendix C	Application data and results	56
Table C.1	Constants for the empirical formulation of the ambient dose $H^*(10 \text{ mm})$ albedo $\alpha_D(E_o, \theta_o, \theta, \phi)$ for neutrons incident on NBS Type 04 concrete	59
Table C.2	Constants for the expressions fitting the Coleman et al. differential and total albedo data for intermediate-energy neutrons incident on reinforced concrete	61
Table C.3	Constants for the empirical formulation of the ambient dose $H^*(10 \text{ mm})$ albedo $\alpha_D(E_o, \theta_o, \theta, \phi)$ for gamma rays incident on NBS Type 04 concrete	62
Table C.4	Parameters for the five-term approximation of the ambient dose equivalent albedo for reflected secondary gamma rays resulting from neutrons incident on ordinary concrete	62
Table C.5	Attenuation factors for monoenergetic beams of gamma rays obliquely incident on slabs of Type 04 ordinary concrete, expressed as the ratio of transmitted to incident air kerma	63
Table C.6	Neutron and secondary-gamma-ray transmission factors for ambient dose $H^*(10)$ (PAR) for neutrons incident on 15-cm-thick slabs of NBS Type 04 ordinary concrete	64
Table C.7	Neutron and secondary-gamma-ray transmission factors for ambient dose $H^*(10)$ (PAR) for neutrons incident on 30-cm-thick slabs of NBS Type 04 ordinary concrete	65
Table C.8	Neutron and secondary-gamma-ray transmission factors for ambient dose $H^*(10)$ (PAR) for neutrons incident on 50-cm-thick slabs of NBS Type 04 ordinary concrete	66
Table C.9	Neutron and secondary-gamma-ray transmission factors for ambient dose $H^*(10)$ (PAR) for neutrons incident on 75-cm-thick slabs of NBS Type 04 ordinary concrete	67
Table C.10	Neutron and secondary-gamma-ray transmission factors for ambient dose $H^*(10)$ (PAR) for neutrons incident on 100-cm-thick slabs of NBS Type 04 ordinary concrete	68
Table C.11	Composition of Hanford ordinary concrete as a function of temperature	69
Table C.12	Compositions of concretes used in measurements	69
Table C.13	ANISN spherical model for pressurized water reactor calculation	70
Table C.14	Material compositions for pressurized water reactor calculation	70
Table C.15	Neutron source distribution for pressurized water reactor calculation	71
Table C.16	Neutron source spectrum for pressurized water reactor calculation	71
Figure C.1	Resonance neutron distribution in ordinary concrete as a function of temperature	72
Figure C.2	Fast neutron distribution in ordinary concrete as a function of temperature	73

Figure C.3	Measured neutron fluxes versus distance in ordinary concrete	74
Figure C.4	Gamma-ray and neutron flux attenuation curves in ordinary concrete	75
Figure C.5	Radiation levels in Bradwell side shield	76
Figure C.6	Dose rates in typical pressurized water reactor shield ...	77
Figure C.7	Gamma-ray dose rates from 4-in. Schedule 80 and Schedule 160 steel pipe, containing airborne or waterborne radioactivity, derived through use of the QADMOD-GP code	78
Figure C.8	Gamma-ray dose rates from 8-in. Schedule 80 and Schedule 160 steel pipe, containing airborne or waterborne radioactivity, derived through use of the QADMOD-GP code	79
Figure C.9	Gamma-ray dose rates from 12-in. Schedule 80 and Schedule 160 steel pipe, containing airborne or waterborne radioactivity, derived through use of the QADMOD-GP code	80
Figure C.10	Gamma-ray dose rates from 16-in. Schedule 80 and Schedule 160 steel pipe, containing airborne or waterborne radioactivity, derived through use of the QADMOD-GP code	81
Figure C.11	Gamma-ray dose rates from 24-in. Schedule 80 and Schedule 160 steel pipe, containing airborne or waterborne radioactivity, derived through use of the QADMOD-GP code	82
Figure C.12	Gamma-ray dose rates from 6-ft outer diameter tank, with 0.25-in. steel wall, containing airborne or waterborne radioactivity, derived through use of the QADMOD-GP code	83
Figure C.13	Penetration types	84
Figure C.14	Geometry for albedos and a sample application	85
Figure C.15	Reflection geometries	86

Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants

1 Scope

This standard contains methods and data needed to calculate the concrete thickness required for radiation shielding in nuclear power plants. Where possible, specific recommendations are made regarding radiation attenuation calculations, shielding design, and standards of documentation. The standard provides guidance to architect-engineers, utilities, and reactor vendors who are responsible for the shielding design of stationary nuclear plants. This standard does not consider sources of radiation other than those associated with nuclear power plants. It also excludes considerations of economic aspects of shielding design.

Concrete is a mixture of materials, the exact proportions of which will differ from application to application. This standard includes a discussion of the nature of concrete, emphasizing those variable aspects of the material that are important to the shield designer. The document discusses methods of analysis and the shielding input data appropriate to each method. Applications of the analytical methods are given, including bulk transport, radiation heating, streaming, and reflection problems.

2 Requirements and recommendations

2.1 Conformance

The word “shall” is used to denote a requirement; the word “should” is used to denote a recommendation; and the word “may” is used to denote permission, neither a requirement nor a recommendation. To conform with this standard, all concrete shield analyses and designs shall be performed in accordance with its requirements, but not necessarily with its recommendations.

2.2 Requirements

2.2.1 Calculational methods

Any applicable method may be used by the designer in the analysis of shield effectiveness. The designer shall be aware, however, of any limitations imposed by the method employed. Approximations shall be chosen such that the attenuation afforded by the concrete shield is known to be conservative with respect to the design objective. Conservatism may also be introduced by other means, such as the source strength used or the radiation design dose rate outside the shield; the concrete shield analysis need not necessarily be inherently conservative.

2.2.2 Data

Selection of material composition, density, cross sections, albedos, or other properties shall be made such that calculational results are conservative with respect to the design objectives as measured by attenuation afforded by the shield.

2.2.3 Operational environment

Nuclear heating shall be considered during the determination of the operating temperature and water content of a concrete primary reactor shield and of any other concrete shields that are exposed to an incident energy flux greater than 10^{10} MeV/cm² s and that will operate at a temperature of 65°C or greater.

2.2.4 Penetrations

All penetration configurations in concrete shield walls shall be shown to provide adequate attenuation. This requirement shall be satisfied by one of the following:

- (1) analysis that follows the guidance of Sec. 8.4 of this standard;
- (2) determination that the configuration is similar to one that is functioning properly under comparable conditions in an operating nuclear facility;

(3) determination that the configuration is similar to one that has been evaluated experimentally and found to be effective for the radiation levels under consideration;

(4) determination that the configuration is similar to one that has already been shown by analysis to be effective.

2.2.5 Reflection

Each applicable reflection configuration shall be reviewed to determine its effect on the design radiation levels. (See Sec. 8.5.3.)

2.2.6 Quality assurance

The quality assurance requirements of American National Standard “Quality Assurance Requirements for Nuclear Facility Applications,” ANSI/ASME NQA-1-2004 [1]¹⁾, Sections 3, 3S-1, and 3A-1, shall be followed. As a consequence of these requirements, the shield designer shall review the initial specification for concrete to be used in shield walls and shall review any subsequent changes to that specification. The shield designer shall ensure the effectiveness of the shield wall based on the reviewed concrete specification. In addition to the quality assurance requirements of ANSI/ASME NQA-1-2004 [1], concrete shielding designed to protect plant personnel should be tested in accordance with American National Standard “Program for Testing Radiation Shields in Light Water Reactors (LWR),” ANSI/ANS-6.3.1-1987 (R1998) [2].

2.3 Recommendations

2.3.1 Calculational methods

The following methods, listed in order of increasing complexity and sophistication, are suitable for determining the effectiveness of a concrete shield wall: the point-kernel method; the one-dimensional (1-D), two-dimensional (2-D), or three-dimensional (3-D) discrete ordinates methods; and the Monte Carlo method, as described in Sec. 6. The following methods, listed in order of increasing complexity, are suitable for determining the effects of a penetration in a concrete shield wall: the albedo

method, the single scatter method, and the Monte Carlo method, as described in Sec. 8.4. The following methods, listed in order of increasing complexity, are suitable for evaluating the effects of reflection from a surface: the albedo method, the single scatter method, and the Monte Carlo method, as described in Secs. 8.4 and 8.5. The adequacy of the method used should be demonstrated for typical shield applications rather than for each shield. This adequacy should be documented and may be based on comparison with experiments, comparison with field measurements, comparison of varying levels of computational sophistication, or other confirmatory comparisons.

2.3.2 Data

In the analysis of a concrete primary reactor shield, a coupled neutron–gamma-ray library may be employed, and the data library recommended in American National Standard “Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power Plants,” ANSI/ANS-6.1.2-1999 [3], is suggested for use. The gamma-ray mass attenuation coefficients and buildup factors that are given in American National Standard “Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials,” ANSI/ANS-6.4-3-1991 (withdrawn 2001) [4], should be used.²⁾ If the actual concrete density and the chemical composition are known, based on concrete mix specifications or measurements of test samples, they should be used. If the chemical composition of an ordinary concrete is not known, the composition of Type 04 concrete as specified in Table 1 should be used.

2.3.3 Other considerations

Neutron activation of trace elements should be considered. Rebar (reinforcing steel) should be considered separately from the concrete in order to allow determination of the sources of secondary gamma rays.

2.3.4 Documentation

A document summarizing the shield design should be prepared at a suitable time in the design evaluation. Chapter 12 of the Final Safety

¹⁾ Numbers in brackets refer to corresponding numbers in Sec. 9, “References.”

²⁾ Other data contained in this standard, including the data in the Appendices, may be used by the shield designer.

Table 1 – Typical concrete properties [5]

Concrete type	Ordinary [6]	Ordinary [7]	Magnetite	Barite	Magnetite and steel punchings	Limonite and steel punchings	Serpentine [6]
Designation	03	04	M	BA	MS2	LS	S
Composition of mix (lb/yd ³)							
Water							
Cement							
Aggregate	(Sand and gravel)	(Sand and gravel)	330 875 4900 (magnetite ore)	370 550 4980 (BaSO ₄ ore)	340 940 1900 (magnetite ore) 4800 (punchings)	347 980 1820 (limonite ore) 4680 (punchings)	373 525 2030 (serpentine) 956 (sand)
Density of cured concrete (theoretical)							
In g/cm ³	2.39	2.35	3.53 ^a	3.35 ^a	4.64 ^a	4.54 ^a	2.1 ^a
In lb/ft ³	151 ^b	148 ^b	220 ^a	212 ^a	290 ^a	283	132 ^a
Elemental composition of cured concrete (g/cm ³)							
H (in mix)	0.02	0.013	0.011 ^a	0.012 ^a	0.011 ^a	0.014 ^a	0.035 ^a
H (in ore)						0.017	
O (in mix)	0.159	0.103	0.087 ^a	0.097 ^a	0.090 ^a	0.091 ^a	1.126
O (in ore and cement)	0.980	1.068	1.081	0.946	0.548	0.617	
Si	0.342	0.742	0.091	0.035	0.073	0.067	0.460
Ca	0.582	0.194	0.251	0.168	0.258	0.261	0.150
C	0.118						0.002
Na		0.040					0.009
Mg	0.057	0.006	0.033	0.004	0.017	0.007	0.297
Al	0.085	0.107	0.083	0.014	0.048	0.029	0.042
S	0.007	0.003	0.005	0.361			
K	0.004	0.045				0.004	0.009
Fe	0.003	0.029	1.676	0.159	3.512	3.42	0.068
Other	Ni: 0.026, P: 0.007		Ti: 0.192, V: 0.011, Mn: 0.007, Cr: 0.006	Ba: 1.551	Ti: 0.074, V: 0.003	V: 0.004	Cr: 0.002
^a Concrete in which 50% of the water added to the original mix is retained.							
^b Minimum acceptable density 140 lb/ft ³ .							

Analysis Report (FSAR) may serve in lieu of a separate document. However, if a separate document is prepared, the guidelines of Sec. 3 of this standard should be followed.

3 Standards of documentation

Documentation that summarizes the shield design, and all related calculations, shall be prepared at a suitable time in the evolution of the shield design for quality control purposes and for future reference. This documentation should include sections listed in Secs. 3.1 through 3.6 of this standard.

3.1 Shield design approach

This section should describe the approach used in the design of the shielding, including appropriate discussions of those portions of the shield design that are based on earlier designs or operating experience.

3.2 Shield design description

This section should contain a detailed description of the shield design, including major equipment locations and shielding arrangements, type of concrete used, shielding thicknesses, major sources of radiation to be shielded, and radiation zone designations throughout the plant. The results of analyses of radiation levels, with the shielding as described, should also be presented either in this section or in a separate section as described in Sec. 3.4. A discussion should also be included that indicates the design phase (e.g., conceptual phase, preliminary phase, etc.) upon which the present shield analyses are based.

3.3 Methods of analysis

The methods of analysis used in design of the shielding should be described in general terms with appropriate discussions of computer programs employed. Sufficient detail should be included to provide for a basic understanding of the theory involved and the specific applications. Examples (drawings or sketches) should be included to depict how specific configurations were modeled for computer representations. These examples should depict boundary and region designations and convey the general level of complexity involved in the analysis.

3.4 Description of analyses

This section may be presented separately or in combination with Sec. 3.2. The material to be included should be subdivided by portions of the plant (e.g., Reactor Building, Auxiliary Building, etc.).

This section may contain dose levels and other shielding criteria used in the design, a description of the sources of radiation to be shielded, the location and extent of shielding, sufficient description of the models employed and simplifying assumptions to allow a specialist in the field to evaluate independently the adequacy of the model and assumptions employed, reference to the methods of analysis used in the specific area, and the calculated radiation levels throughout the nuclear plant and at the boundary of the unrestricted area surrounding the plant. Basic data used in the analyses should also be included or referenced (e.g., attenuation coefficients, buildup factors, etc.).

3.5 References

A list of references should be included with particular emphasis on referencing computer programs. Appropriate reference should also be made to the origin of sources, assumptions, etc., whenever these are documented elsewhere (e.g., FSAR sections or design-basis documents).

3.6 Summary results

This section should contain a brief summary of the shielding design and analysis, showing that the objectives are met.

4 Terms and definitions

The following terms and definitions are provided to assure uniform understanding of selected terms as they are used in this standard. Additional terms are defined in the following documents: *Glossary of Terms in Nuclear Science and Technology* [8], and the American Concrete Institute's "Cement and Concrete Terminology," ACI-116R00 (R2005) [9].

A number of terms are used in this standard as defined in several textbooks on the subject of radiation shielding. They are not used in any special sense in this standard.

aggregate: The granular material that is mixed with water and cement to form concrete. Concrete for radiation shielding is classified as ordinary or high density according to the unit weight of aggregate used. The aggregates defined in this section are those usually used for radiation shielding; other aggregates are sometimes used when it can be shown objectively that they produce concrete of the required strength, durability, and shielding characteristics.

aggregate for ordinary concrete: Fine and coarse aggregates described in “Descriptive Nomenclature of Constituents of Concrete Aggregates,” ASTM C294-05 [10]. They must normally meet the requirements of “Specification for Concrete Aggregates,” ASTM C33-03 [11], except aggregates for concrete to be placed by the preplaced aggregate method that meet the gradation requirements of “Guide for the Use of Preplaced Aggregate Concrete for Structural and Mass Concrete Applications,” ACI 304.1R-92 (R2007) [12].

aggregate for high-density concrete: Special aggregate, or mixtures of special aggregate with the natural mineral aggregates used for normal density concrete. Special aggregates are iron shot, steel punchings, or natural mineral or synthetic aggregate as described in “Descriptive Nomenclature of Constituents of Aggregates for Radiation-Shielding Concrete,” ASTM C638-92 (R2002) [13], and meeting the requirements of “Specification for Aggregates for Radiation-Shielding Concrete,” ASTM C637-98a (R2003) [14].

hydrous aggregate: Special aggregates with a relatively high water-of-hydration content, which may be used if the water content of the cured concrete cannot be maintained at a level high enough for the desired neutron attenuation requirements with the aggregates specified in the two previous paragraphs. Properties of hydrous aggregates are given in ASTM C637 [14].

boron-containing aggregates: Aggregates that contain boron, which are sometimes mixed with other aggregates to enhance neutron capture and reduce secondary-gamma-ray production. Boron-containing minerals are described in ASTM C638 [13].

absorbed dose: The energy imparted by radiation per unit mass of irradiated material (100

erg/g = 1 rad absorbed dose; in SI units, 1 J/kg = 1 Gy absorbed dose).

albedo: The probability under specified conditions that a particle entering into a region through a surface will return through that surface.

analysis: The examination of radiation transport in and through shields, and the determination of its effects, including predictions of dose rates and neutron and gamma-ray fluxes as modified by the introduction of shields in the systems involved.

design: Technical and management processes that lead to and include the issuance of design output documents such as drawings, specifications, and other documents defining technical requirements and performance of radiation shields.

dose: In this standard, taken to refer to “dose equivalent,” the product of the absorbed dose, the quality factor, and other modifying factors at the location of interest.

grout: A fluid mixture of cement, water, and possibly some fine aggregate.

neutrons

fast neutrons: Neutrons of kinetic energy greater than some specified value. This value can vary over a wide range and will be dependent upon the application. In concrete shield analyses, the specified value is normally 0.1 MeV in the calculation of tissue doses and 1 MeV in the determination of radiation damage to materials.

intermediate-energy neutrons: Neutrons with greater energy than thermal neutrons and less than the energy associated with fast neutrons. This range also includes the resonance neutron energy range in which many nuclides exhibit strong neutron absorption, referred to as resonance absorption.

thermal neutrons: Very low-energy neutrons that are in thermal equilibrium with the atoms (or molecules) of the medium in which they are present. They have a Maxwellian distribution of speeds, with the most probable speed being 2200 m/s in a medium at 20°C (corresponding to a kinetic energy of 0.025 eV).

ordinary concrete: Type 04 concrete having a theoretical density of 2.35 g/cm³. This type of

designation is based on the elemental composition as defined in “A Summary of Shielding Constants for Concrete,” ANL-6443 [15]. There are many different types of ordinary concrete; the differences are due to variation in the mix proportions and elemental composition of the cement, sand, coarse aggregate, and water. Reference [15] describes Type 04 ordinary concrete and other types of ordinary concrete as well.

rebar: The reinforcing steel bars placed in concrete to increase the structural strength. “Code Requirements for Nuclear Safety Related Concrete Structures and Commentary,” ACI 349 [16], includes a discussion of this reinforcing steel. Owing to the fact that it is not homogeneously spaced throughout the shield, but rather is made up of discrete steel bars, it would require complicated shielding analysis and hence is not usually considered as part of the attenuating property of the shield. Rebar is, however, considered as a potential source of secondary gamma radiation when the shield is a combined neutron and gamma-ray shield.

reflection: A process in which radiation enters a region through a surface and partially returns through the same surface.

streaming: The penetration of radiation through gaps, voids, or ducts existing in the shield structure.

crete is one of the most economical means of placing bulk material around a radiation source in a stationary nuclear power plant, although concrete’s low density, relative to most metals, generally means a sacrifice in terms of space required. Concrete is valuable as a fast neutron shield, as it contains water as a constituent part. The hydrogen nucleus is uniquely effective in slowing down fast neutrons.

Concrete usually is mixed on or near a job site. The primary constituents of ordinary concrete, in terms of their partial densities in the mix, usually fall in the ranges shown in Table 2 (see Ref. [17]). Conversion factors for commonly used construction specifications are also shown.

Newly mixed, or fresh, concrete is a viscous fluid and normally is poured into forms constructed to produce the desired shape of the finished structure. Carbon steel reinforcing bars in the required number and pattern are fixed within the forms prior to the addition of the fluid concrete. After placement, the concrete cures through a process in which the cement and a portion of the mix water react exothermically (hydration). As concrete cures, its compressive strength increases. The initial rate of hydration is high, and most concretes reach 50% or more of their ultimate strength in about seven days. Completion of curing, to 90% of ultimate strength, depends on the type of cement used and may require several months.

The exact proportions of the materials used to mix concrete for a particular application depend upon the specific details of the application. (See Chap. 2 of Ref. [18]). In general, the amount of cement used is proportional to the desired strength of the material. The amount of water included in the mix depends on the combined requirements for hydration and on the specified consistency and workability of the

5 Characterization of concrete

5.1 Introduction

Concrete is one of the most versatile and widely used structural materials. In cases where protection from penetrating neutrons and gamma rays is a design consideration, this material is used extensively for shielding. The use of con-

Table 2 – Partial densities of concrete constituents

Ingredient	Partial Density (g/cm ³)	Construction Unit Conversion Factors (g/cm ³)
Portland cement	0.2 to 0.5	1 bag/yd ³ = 5.58×10^{-2}
Water	0.13 to 0.25	1 bag/yd ³ = 4.95×10^{-3}
Aggregates (sand and gravel)	1.2 to 1.9	1 bag/yd ³ = 5.94×10^{-4}

fresh concrete mix. The latter consideration governs how well the mix can be handled and usually is the determining factor for the water content. (See Ref. [19]). Most (45 to 85 vol%) of the bulk of concrete is aggregate. The amount of aggregate and its size gradations (coarse to fine) are determined by many factors, including the desired workability of the mix and the specified strength and durability of the finished concrete. By changing the materials in the aggregate, the radiation shielding effectiveness of the concrete can be altered (e.g., through the use of high-density aggregate materials).

This brief discussion of concrete technology is provided, in part, to illustrate the degree of variability in properties that can exist in concrete. In some cases, the variations can be controlled (aggregate density); in others they are determined by the nature of the physical process (water content) or are produced inadvertently (voids in poured concrete). The remainder of Sec. 5 contains a more detailed discussion of the major factors that can affect the properties of concrete as a radiation shielding material. The construction aspects of concrete structures within nuclear power plants are addressed in ACI 349-01/349R-01 [16].

5.2 Concrete placement

The placement of concrete is extremely important in assuring that its shielding effectiveness will be as designed (see Chap. 5 of Ref. [16] and Chaps. 24, 25, 26, and 38 of Ref. [18]). Once poured, or placed, it is difficult to evaluate the quality of a concrete shield, short of actual operational testing (see Ref. [19]).

In conventional placement the fresh concrete is poured, or chuted, into the forms. Precautions must be taken in this operation to ensure that the coarse aggregate does not separate from the mix. This problem is more acute when heavy aggregates are used and the specific gravity of the aggregate particles is much greater than that of the mix. Another potential problem in conventional placement is the creation of voids in the poured material. This situation can arise where there are many closely spaced reinforcing bars, or many embedments. It is extremely important to specify that the maximum size of the coarse aggregate be smaller than the characteristic spacings between reinforcing rods or embedments. Fresh concrete generally is vi-

brated or rodded after placement to assure the filling of spaces between obstructions.

Preplaced aggregate is a method of concrete placement in which the forms are first filled with the graded, coarse aggregate. A grout is injected into the aggregate-filled forms, usually from the bottom, filling the spaces between the aggregate particles and thus creating concrete. This type of placement is useful with a heavy aggregate that would be difficult to handle with conventional equipment or when the space or geometric restrictions of the shield make it difficult to assure that poured concrete could be placed to fill all voids.

Another method of concrete placement is the use of precast blocks. These can be stacked to create a shield of any desired configuration. This method is particularly valuable if there is a possibility that the shield wall might have to be removed. Blocks can be made of ordinary or heavy aggregate concrete and have the advantage that they can be tested after forming for proper density.

5.3 Water content

See Refs. [6], [17], and [20] for an excellent review of the water content of concrete. Most of this summary is taken from these references.

5.3.1 Significance of water content

As mentioned in Sec. 5.1, the water (or hydrogen) content of cured concrete is important in the determination of its neutron shielding characteristics. (See Ref. [21].) It is sometimes difficult to obtain a quantitative measure of the actual water content of as-poured concrete, and in fact, this value can change during the lifetime of the structure. In many reactor shielding situations, the secondary gamma radiation produced within the primary shield is a more important contribution to the dose outside the shield than is the neutron radiation. Thus, variations in the neutron thermalization properties of concrete might be of concern as they affect the spatial distribution of secondary-gamma-ray production.

5.3.2 The nature of water in concrete

Water is added as a constituent of the concrete mix. As the hydration reaction proceeds, free water from the mix becomes chemically bound to the hydrated cement. The chemically reacted, or bound, water is fixed in the concrete

and cannot be lost by evaporation. The water remaining in the pore system of the cement-water paste acts as a source of reaction water during the hydration period. This water is often termed “free water.” It can diffuse through the concrete, and under conditions of high concrete temperature or extreme dryness, or both, at the exterior of the concrete, much of it will be lost. It can, in part, also be replaced if the concrete surface is subsequently exposed to high humidity or moisture.

Water can also be present in the aggregate pores and behaves much like the free water of the mix. In general, this source does not represent a significant portion of the concrete water content. Water that is chemically bound within the aggregate behaves as the hydrated water and is a stable component of the water content of the concrete, even under extreme temperature and environmental conditions. Only the hydrous ores (serpentine and limonite) have appreciable water contents; these aggregates may be specified in situations in which concrete with high water content is required.

5.4 Heating effects

In the design of a concrete radiation shield, it is necessary that the temperature and temperature distribution throughout the shield be calculated prior to construction. In addition to radiation heating sources, these calculations must include detailed consideration of other heat sources and sinks. Although structural considerations are outside the scope of this standard, the shield designer should be aware that thermal changes resulting from the radiation environment may affect the ability of concrete to meet its structural requirements. (See Sec. 8.1.2.)

The primary impact of temperature on the attenuation characteristics of concrete is associated with the loss of hydrogen or water from the system. Drying of concrete can be divided into three time periods. The first two are relatively short and depend upon air velocity at the surface of the concrete and if the surface is relatively wet or moist. During the third drying time period, the rate depends on the properties of the concrete and the environmental conditions. Drying begins at the exposed surface and extends slowly inward. This is an evaporation process that is fed by water in the pore system of the concrete. A continuous moisture transfer occurs until equilibrium is reached.

The rates and equilibrium conditions are established by the water-cement ratio and degree of hydration; by the relative humidity at the surfaces; and by the temperature of, and temperature distribution within, the concrete.

5.5 Reinforcing steel

Concrete has significant structural strength only in compression. Its tensile strength is negligibly small and must, in all cases, be provided by steel bars, embedded in those portions of the structure that are subjected to tensile loads. The resulting so-called “reinforced concrete” derives its strength from the steel acting together with the encasing concrete to resist applied forces.

When the density and elemental composition of a concrete section are calculated for shielding purposes, the purpose of the calculation shall determine how the rebar will be considered. For gamma-ray sources, the presence of rebar in concrete may be considered, and if considered, a simple homogenization of the rebar and concrete is sufficient to account for its effect. For neutron sources, the presence of rebar in concrete may be considered, but if it is, a simple homogenization is not adequate. Instead, the localized regions of high steel concentration should be considered as one or more separate shield layers, each different in density and element composition from the bulk, unreinforced portion of the concrete. Such a heterogeneous treatment may be required because of two competing effects caused by the rebar: The production of neutron-induced gamma-ray sources is increased, but the gamma-ray attenuation is also increased. The effects of these two mechanisms on actual calculations are discussed further in Sec. 6.3.

The exact location and size of rebar in a particular shield can be obtained from the construction drawings. In most cases, reinforcing steel will be a low-carbon steel, which can be adequately represented as pure iron for most shielding and secondary capture gamma-ray source purposes. However, if the secondary-gamma-ray production in the rebar is expected to be an important source term, the exact composition of the steel should be determined.

5.6 Aggregates

It is through the aggregate that the properties of concrete can be modified and controlled most

readily for the purpose of enhancing its shielding effectiveness. The most common procedure is the use of a metallic ore as the coarse aggregate to increase the density and effective atomic number of concrete. Certain aggregate minerals contain bound water and can be used to assure a high hydrogen content in the shield. Also, aggregate materials that are rich in the neutron absorber boron can be used to reduce the intensity of secondary-gamma-ray production from thermal neutrons. (See Chap. 9 of Ref. [6], and also see Ref. [13].)

Table 1 provides a summary of the properties, including the elemental partial densities, of concrete formed through the use of various aggregates. The source of Table 1 is Ref. [5]; similar data for the former Soviet Union are available in Ref. [22]. The properties for Type 03 ordinary concrete and serpentine concrete are from Ref. [6], and those for Type 04 ordinary concrete are from Ref. [7]. The rest of this section is devoted to a discussion of some of the different kinds of aggregates.

5.6.1 Heavy aggregates

There are two principal types of heavy aggregate used to create dense concrete: heavy elemental ores, and iron or steel punchings. They can be used as coarse aggregate, ground to form fine aggregate, or used in combination with each other. Characteristically, heavy aggregates present problems in the mixing and handling of concrete. The mix densities can be several times that of normal concrete, and the heavy constituents have a tendency to settle out of the mix. Special placement techniques, such as are mentioned in Sec. 5.2, are sometimes necessary. Some of the naturally occurring materials that can serve as heavy aggregates are listed in Secs. 5.6.1.1 through 5.6.1.4.

5.6.1.1 Ilmenite (FeTiO_3 , with minor Mg and Mn)

Ilmenite has a hardness of 5 to 6 and specific gravity of 4.72 ± 0.04 in the pure mineral.

5.6.1.2 Limonite

Limonite is a general name for hydrous iron oxides of unknown composition, frequently cryptocrystalline goethite (HFeO_2) with adsorbed and capillary water, and probably mixtures of such goethite with similar lepidocrocite ($\text{FeO}(\text{OH})$) or hematite (Fe_2O_3) or both, with

adsorbed and capillary water. Specific gravity ranges from 2.7 to 4.3.

5.6.1.3 Magnetite (Fe_3O_4)

Magnetite has a hardness of 5.5 to 6.5 and specific gravity of 5.17 in the pure mineral. Magnetite ores can form dense, tough, usually coarse-grained rocks with few impurities; magnetite ores are associated with metamorphic or igneous or sedimentary rocks, and therefore, the impurities associated with magnetite ores can include a wide variety of rock-forming and accessory minerals.

5.6.1.4 Barite (BaSO_4)

Barite (also called barytes) has a hardness of 3 to 3.5 and a specific gravity of 4.50 in the pure mineral.

5.6.2 Hydrous ores

Hydrous ores contain water in a bound state, which will not be driven from the concrete even at high temperatures. Under high-temperature conditions that might reduce the water content of ordinary concrete to the minimum value of that bound in the cement (see Sec. 8.2), it is desirable to use a hydrous aggregate. Two of the common hydrous ores, limonite and goethite, contain 7 to 12 wt% of water, which will remain bound to temperatures up to 200°C.

From the standpoint of water quantity and high-temperature retention, the most attractive aggregate is serpentine rock. (See Ref. [23].) This mineral has a fairly extensive distribution throughout the world. Serpentine is an asbestos mineral, a hydrous magnesium silicate with the formula $3\text{MgO} \cdot 2\text{SiO}_2 \cdot 2\text{H}_2\text{O}$. The water of hydration in serpentine amounts to ~ 13.5 wt%. The rock has a lump density of between 2.55 and 2.65 g/cm³ (160 to 165 lb/ft³). Results of tests conducted on concrete made with serpentine rock as the aggregate indicate that not only is the hydrated water held at temperatures as high as 400 to 500°C, but also when mixed in the correct proportions, the concrete has very good strength. When dried to constant weight at the temperature of 43°C, the concrete has a density of ~ 2.09 g/cm³ (130 lb/ft³).

5.6.3 Boron-containing minerals

The gamma rays that result from neutron capture by the isotope ^{10}B are much less penetrating than those that result from neutron capture

by hydrogen or other elements. For this reason, boron and boron compounds may be used in neutron shields. The unusually high capture capability of ^{10}B permits its use in relatively small quantities. Boron is most frequently incorporated in the concrete as borate minerals or synthetic boron frits. Both methods cause some retardation of set of the concrete.

The experience recorded in the United States suggests that the cost of synthetic boron frits, which is higher than the cost of borate, can be counterbalanced by uniform composition of the frits, which permits effective control of the properties of the concrete.

Borate production in the United States is virtually limited to borax ($\text{Na}_2\text{B}_4\text{O}_7 \cdot 10\text{H}_2\text{O}$) and borax derivatives obtained from natural brines at Searles Lake, California, and brines produced by treating borates from the Kramer deposit at Boron, California. California colemanite ($\text{Ca}_2\text{B}_6\text{O}_{11} \cdot 5\text{H}_2\text{O}$) deposits are apparently not regularly worked, but colemanite ores have been obtained from them for use in shielding concrete.

Boron minerals that are stable and insoluble are usually not available in large quantities for use as aggregates. The recorded exceptions are Paigeite (Vonsenite) ($\text{Fe}_2^{2+}\text{Fe}^{3+}\text{BO}_5$) and Tourmaline ($\text{Na}(\text{Mg}, \text{Fe}, \text{Mn}, \text{Li}, \text{Al})_3\text{Al}_6\text{Si}_6\text{O}_{18}(\text{BO}_3)_3(\text{OH}, \text{F})_4$).

6 Calculation methods

6.1 Introduction

Various methods are available to the shield designer for calculating the radiation field outside concrete shields. Some of the techniques, mainly kernel methods, are simple enough to allow computation by hand; however, computerized calculations are preferable and are readily performed. The discrete ordinates and Monte Carlo approaches must be carried out by digital computer since they require thousands of repetitive calculations.

The decision of which technique to use is not always an obvious one. If some of the simpler methods can be justifiably applied, they should be used first. In the case of nuclear power plants, kernel methods are generally applicable, although there are several notable exceptions where more sophisticated methods are needed

to better represent the radiation transport involved. These are the primary shield and its reactor sources, and areas involving scattering problems, e.g., labyrinths, ducts, piping penetrations, and “skyshine” situations.

6.2 Point-kernel methods

Kernel methods have the merit of simplicity, so that even when a more accurate calculation is needed, they can often be used as a check. The basis of the method is the physically reasonable premise that the radiation field resulting from a distributed source can be obtained by breaking it up into many smaller sources and evaluating and summing each of their contributions.

The kernel $K(P' \gg P)$ is the function that describes how the detector point P is influenced by the unit source point at P' . The point kernel is the basic approach that is used, making the kernel a simple function of the distance R separating P and P' . In an infinite homogeneous medium, the kernel for the uncollided radiation is $K(P' \gg P) = (4\pi R^2)^{-1} \exp(-\mu R)$, where $\mu = \mu_t(E)$ = attenuation coefficient of the medium and E = radiation source energy. This kernel gives the particle flux at P due to a point isotropic source at P' emitting one particle per unit time.

The total radiation field, both scattered and unscattered, is desired since it is either the radiation exposure (dose rate) or the energy deposition (which depends on the total radiation flux) that is generally the quantity of interest. To transform from particle (either photon or neutron) flux to dose rate, dose rate conversion factors are available, such as those tabulated by the National Council on Radiation Protection and Measurements (see Ref. [24]), the International Commission on Radiological Protection (see Ref. [25]), and American National Standard “Neutron and Gamma-Ray Fluence-to-Dose Factors,” ANSI/ANS-6.1.1-1991 (withdrawn) [26]. For photons the simple form of the kernel can be maintained by introducing an empirical factor, called the buildup factor $B(\mu R, E)$, to account for the scattered radiation. Thus, the infinite homogeneous medium point kernel is $K(R, E) = B(\mu R, E)(4\pi R^2)^{-1} \exp(-\mu R)$.

Since the buildup factor is introduced as a tool of convenience, its identification allows a response function to be incorporated to show the effect of scattering on the quantities of inter-

est. Thus, radiation buildup factors have been defined for radiation exposure and energy deposition. Graphs, tables, and empirical fitting functions of the buildup factors for concrete are given in Sec. 7.

The point-kernel approach focuses on the distance between source point and detector in terms of mean free paths (mfp), i.e., the product of attenuation coefficient and path length as the measure of attenuation through material. From this standpoint, then, the kernel can be generalized to include path lengths through several shield materials:

$$K(R, E) = (4\pi R^2)^{-1} B\left(\sum_i \mu_i t_i, E\right) \times \exp\left(-\sum_i \mu_i t_i\right),$$

where:

t_i = distance traversed through material i in going from P' to P ;

μ_i = corresponding attenuation coefficient.

When there is more than one shield, the buildup factor $B(\sum_i \mu_i t_i, E)$ can still be defined, but to express it in terms of properties of the individual shield materials is more difficult. For this reason, some simplifying guidelines that can be followed are available (see Chap. 3.2 of Ref. [27] and Ref. [28]), although some analytical work (see Refs. [29], [30], and [31]) has attempted to deal more rigorously with the multiple-layer buildup factor. In practice, it is preferable to evaluate the buildup factor in terms of the total number of mean free paths, ascribing that total number to one representative material, making certain the selected material is conservative for the given configuration.

The photon point-kernel technique is employed in two general ways: for simplified analytical solutions and for direct numerical integration. First, it can be integrated over idealized source geometry configurations to provide simplified equations for the uncollided flux. These integrations may be analytical, numerical, or semi-empirical, and they have been widely used in shield design work but are now generally applicable when calculations are performed by hand. Frequently, special functions are re-

quired in the formulas. The pioneer volume edited by T. H. Rockwell [32] lists many of these formulas and their derivations along with tabulations of the required functions. Other sources published after the volume edited by Rockwell include Refs. [27], [30], [31], and [33].

In addition, similar work has been carried out for special cases. One useful approach for cylindrical and spherical volume sources is that of Ono and Tsuruo [34] since it extends the range of source configuration parameters, relative to the method in Rockwell [32], to which the technique can be applied. These source shapes are frequently encountered in power plant shielding. Thus, this versatile approximate solution can be used to obtain upper and lower bounds for complex source and shield arrangements. (See Refs. [35] and [36].)

The second basic use of the kernel is for the direct numerical integration by digital computer codes. These allow detailed descriptions of the source, shield, and adjacent geometry. Many point-kernel codes³⁾ are available from the Radiation Safety Information Computational Center (RSICC), and the RSICC's abstract compilation (see Ref. [37]) lists them along with their relevant operating requirements.

In using any of these codes, it is necessary that the user understand the operation and limitations of the code in order to apply it correctly to the actual source-shield configuration. For most distributed source problems, the volumetric source strength is a key parameter that should be preserved. The maximum size of a subdivided section of a nongaseous source volume generally needs to be on the order of 1 mfp of the dominant source energy group. For dose points close to the edge of the source, a more stringent mesh spacing may be necessary. It is particularly in such cases, which extend the point-kernel method to its limits, e.g., the aforementioned case of close detector proximity (see Ref. [38]) or situations involving potential short circuit paths, that the greatest care should be exercised by the user (see Ref. [39]). Because of the advances in computer technology, it is advisable to use smaller-size source subsections than the maximum noted above.

³⁾ See Appendix A.

Neutron point response kernels have also been developed, but they are subject to various restrictions, and although they are somewhat artificially derived, they can be very useful. The basis of the neutron response kernel is the removal cross section Σ_R , which is not a cross section in its classical sense. Rather, it represents the average attenuation of a great many neutrons suffering numerous interactions of various kinds. Formally, the removal cross section is a measure of the change in neutron dose at a point deep in water when some additional material is placed around the source without altering the original configuration.⁴⁾

Removal theory provides the basis for a point-kernel method for fast neutrons but is now largely of historical interest because of advances in computer technology and widespread use of the discrete ordinates and Monte Carlo methods.

The removal theory concept applies only to fast neutrons and in fact is postulated on a neutron fission spectrum. It can be explained by the argument that some collisions effectively “remove” neutrons either through a large energy loss or change in direction (mainly inelastic scattering and hydrogen collisions). As the lower-energy neutrons are attenuated, the more energetic uncollided neutrons dominate and decrease almost exponentially, like a monoenergetic group. In actual applications it has been found that the removal concept can be applied to several layers of materials, i.e.,

$$D_2(z) = D_1(z) \exp\left(-\sum_i (\Sigma_R t_i)\right),$$

where:

$D_1(z)$ = dose through a distance z of water;

$D_2(z)$ = dose for the same water depth plus the composite of t_i thickness of i materials.

Furthermore, it is also applied to configurations of hydrogenous media other than water (such as concrete).

⁴⁾ Values of Σ_R for concrete are given in Table B.10.

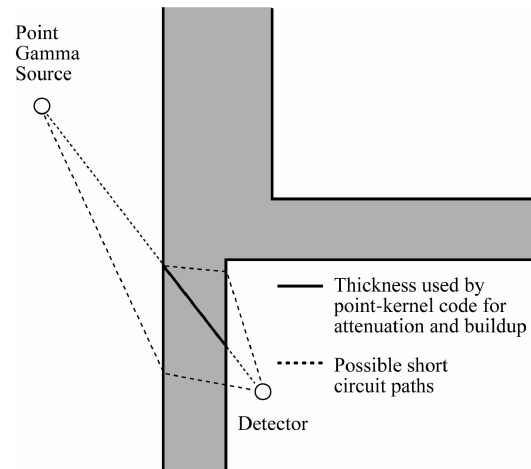


Figure 1 – Example of oblique penetration short circuit paths

The removal theory concept has been extended by its coupling with diffusion theory, as noted in the Spinney Method (see Chap. 5.2.2.2 of Ref. [27]). This method is not recommended since the discrete ordinates and Monte Carlo methods are readily available.

A concluding note of caution is in order regarding the use of the kernel method. It is basically a ray analysis technique, so that if there are streaming paths or “short circuit” paths, the kernel can significantly underestimate the dose rates. Special consideration should also be given to cases of oblique penetration, such as in Fig. 1.

Either special buildup factors should be used (see Ref. [40]), or corrections should be applied to the dose rates (see Ref. [41] and Tables C.5 through C.10 of this standard). Furthermore, in the application to multilayered shields with irregular configurations, there are likely to be deficiencies because these conditions are so far removed from the underlying assumptions. Attenuation and buildup associated with radiation obliquely incident on concrete shields may be treated using attenuation factors or transmission factors such as those tabulated in Appendix C. Finally, with geometries in which scattered surfaces are important, the direct application of buildup factors should not be made unless each such surface is accounted for separately.

6.3 Discrete ordinates method

The Boltzmann equation is the fundamental equation that describes the migration of radiation, neutrons or photons, through media. It is a detailed expression of the balance of particles that is maintained as they traverse a medium; at each point, the particle loss mechanisms equal the production mechanisms. For steady-state conditions it can be formulated in terms of an angular number flux density, dependent on six variables: three spatial variables, two orientation (solid angle) variables, and energy. For most shielding applications, angular flux, $\gamma(x, y, z, \theta, \phi, E)$, is too complex a quantity, and less specific quantities like the scalar flux, $\Psi(x, y, z, E)$, representing an integration of the former over the solid angle, are more useful. The scalar flux is of interest because at a given energy E , it can be weighted with appropriate response functions to provide the quantities desired (e.g., dose rate or energy absorption).

The discrete ordinates method is basically an approach to making discrete the variables of the integrodifferential form of the Boltzmann equation to solve for the angular flux. The 3-D form of the Boltzmann equation, and the 1-D and 2-D reduced forms, have been developed into working codes.⁵⁾ Often, the 1-D or 2-D form is sufficient because of symmetry and other considerations.

The discrete ordinates method is based on several important assumptions: The energy range is divided into energy groups; the angular flux is defined at specified spatial points and energy groups along certain discrete angular directions; the scalar flux is constructed from the angular fluxes using a set of weights matched to the discrete directions; and the complex scattering kernel describing the probability of scattering from one energy and directional state to another is expanded, generally in terms of Legendre polynomials. Further details can be found in Lathrop's comprehensive review [42] as well as in the earlier workshop review [43] and in the documentation of specific codes.⁵⁾

For shielding calculations a fixed source is supplied by the user along with such quantities as

the spatial description, the set of discrete angles and weights, and the energy group cross-section data. The number of angles used in the calculations is described in terms of the quadrature order N and is of such relevance that the method is referred to as S_N .

The cross-section data are dependent on such factors as the type of particles involved (neutron, gamma ray, or coupled neutron–gamma ray), the energy group structure, the order of scattering (number of Legendre polynomial terms used in the expansion), and the number of different materials. These data usually are not prepared by the user but are obtained from information centers like RSICC. Such a data set is very extensive because it describes the scattering kernel by specifying transfer functions to each group from every other group. It thus comprises a nearly square array, G by $G + N$ for each material, where G is the number of energy groups and N represents the additional cross-section data for total, fission, absorption, and similar interaction probabilities.

For analysis of the primary shield of a nuclear plant, a coupled neutron–gamma-ray library can be used to account for the secondary gamma rays created in the shield as well as the source neutrons and gamma rays. It is the coupled cross sections that enable a single calculation to perform what would otherwise require a series of separate neutron and gamma-ray calculations, along with a process for creating the secondary-gamma-ray source terms. The coupled library is thus a combined set of neutron and gamma-ray cross sections containing a scattering matrix that allows for transfer from neutron groups to gamma-ray groups. This transfer accounts for (n, γ) and $(n, \gamma-x)$ reactions such as radioactive absorption, inelastic scattering, and charged particle reactions, as well as prompt gamma rays from fission.

The standard ANSI/ANS-6.1.2-1999 [3] specifies neutron and gamma-ray cross sections and related group-averaged or derived data in the energy range of importance in radiation protection and shielding calculations for material of interest in nuclear power plants. ANSI/ANS-6.1.2-1999 recommends specific coupled neutron–gamma-ray cross-section libraries, and these

⁵⁾ See Appendix A.

are endorsed by this standard as the data sets to be utilized in multigroup analyses.

The importance of the secondary-gamma-ray component also emphasizes that the question of how to model the reinforcing steel within the primary shield should be considered. The available literature on the effect of rebar on the transmitted secondary gamma radiation resulting from an incident neutron flux is limited. (See Refs. [44] and [45].) Measurements and calculations made by the discrete ordinates code (see Ref. [46]) have indicated that the effect is relatively small, i.e., a <20% increase, and, under certain circumstances, even a decrease. The location of the rebar sections within the concrete relative to the incident neutron flux is of prime importance. Rebar within the first few inches of the thermalization zone of the concrete acts mainly as an increased gamma-ray source region, whereas rebar deep within the concrete acts essentially as a region of increased attenuation.

A related but separate effect that should be evaluated for the primary shield is neutron activation. As a result of neutron interactions, nuclides are transmuted into radionuclides that serve as a radiation source even when the reactor is not operating. It is mainly during shutdown that this is an important source.

The long-lived, gamma-emitting isotopes generated within the concrete by neutron activation arise mainly from trace elements. Of the ten elements listed in Table 1 as comprising Type 04 concrete, only four have isotopes with half-lives >1 h: ^{24}Na , ^{42}K , ^{59}Fe , and ^{47}Ca . Over the first few days, ^{24}Na is dominant, but because of its 15-h half-life, it decays away, and the longer-lived but weaker ^{59}Fe remains. Since other types of concrete have different trace elements and in varying proportions, activation products and their resultant radiation fields should be evaluated individually.⁶⁾

In preparing the input data for an S_N calculation, the user should assure that the following guidelines are satisfied. The mesh spacing should not be more than 1 mfp in the vicinity of strong flux gradients. Guides for the quad-

rate order and the order of scattering are that P_3 scattering at a minimum should be used (although for neutrons P_1 may suffice) and that the quadrature order should be at least twice the order of scattering. Thus, while an S_6P_3 combination might typically be used, it is more common to specify S_8P_3 . In the analysis of very thick concrete shields for very high-power reactors, it may sometimes become necessary to consider P_8 scattering. Illustrating the necessity of adhering to these guidelines for fast neutrons in deep penetration problems is Fig. 2, which shows the fast neutron dose rate in water as a function of the order of scattering. (See Refs. [43] and [47].) Through 4 ft of water, the P_0 dose rate is too low by a factor of 10.

The 1-D S_N codes can be beneficially employed in analyzing the reactor primary shield or the leakage spectra exiting thick steel-walled equipment containing radioactive steam (see Ref. [48]). When more detailed information is required, the 2-D S_N codes⁷⁾ can be used to provide the additional axial variation (r - z mode) or azimuthal variation (r - θ mode). The (x - y) mode is also available, but possible power plant applications have emphasized the (r - z) and (r - θ) modes. (See Refs. [49] and [50].)

While the 2-D code will provide more information, namely, separate polar and azimuthal angular fluxes and scalar fluxes throughout the specified 2-D region, it concomitantly requires more computational time and computer storage compared with a 1-D code. In addition, the 2-D code introduces a defect of the discrete ordinates method itself, termed the ray effect, that is absent from the 1-D formulation.

The analytic Boltzmann equation accounts for radiation streaming in all directions, but in its discrete 2-D S_N formulation, the streaming is followed only along specific directions. With isolated sources some regions sometimes cannot receive any source particles, while others, intersected by the allowed directions, sometimes receive too many. The resulting flux oscillations are termed "ray effects"; however, as Lathrop notes [42], these oscillations

⁶⁾ Computer codes that calculate activation are listed in Appendix A.

⁷⁾ See Appendix A.

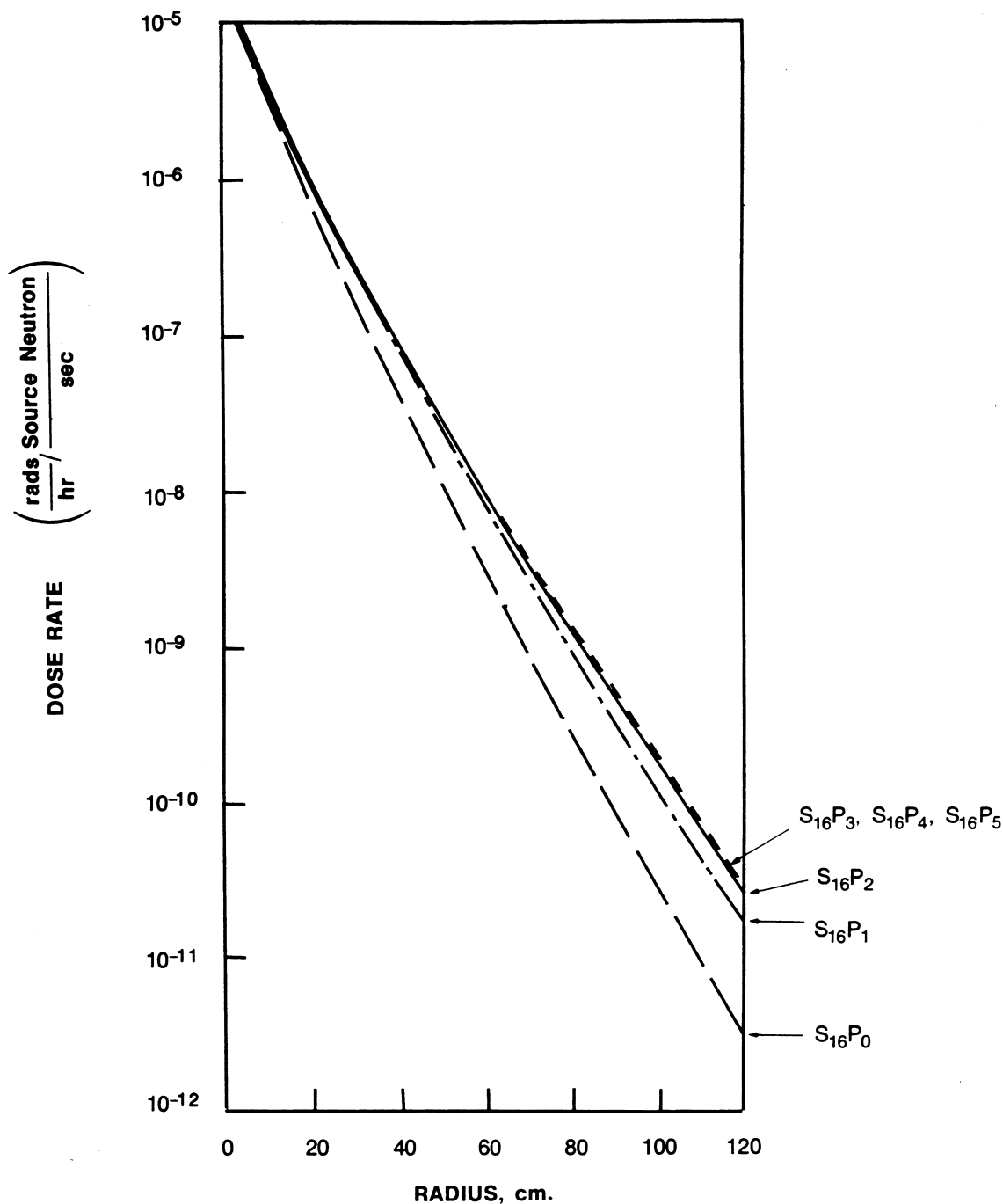


Figure 2 – P_1 convergence of the fast neutron dose rate in water from a point fission source [43, 47]

generally fluctuate about the correct solution. With the proper attention to this and other input considerations, very useful results such as the radiation streaming around the primary shield (see Ref. [49]) can be obtained with 2-D codes.

Diffusion theory is a step in the direction of simplification of the Boltzmann equation since it eliminates angular variables entirely. It is similar to the S_N method in that the flux is solved for group by group, but here it is the scalar flux rather than the angular flux.

Differences that can be expected from the use of diffusion theory rather than transport theory in concrete are shown in Fig. 3. Here, neutron and secondary gamma dose rates based on a fast neutron source are illustrated for the two methods, and diffusion theory underestimates both dose rates. In addition, the gamma dose rate based on a high-energy gamma-ray source is shown for the two methods, and beyond 15 in. of concrete, diffusion theory again underestimates the dose rate, even more strongly than for the previous quantities. These dose rates were calculated with the ANISN code (S_8 and diffusion modes) in cylindrical geometry through the use of the CASK library (see Ref. [51]) to represent Type 02-B concrete (see Ref. [52]).

Diffusion theory can be used to advantage for shielding calculations when it is coupled with removal or transport theory, which treats the deep penetration of fast neutrons more accurately by allowing them to be forwardly directed. The removal-diffusion method considers two different mechanisms and associated energy groupings—removal bands for the high-energy neutrons that serve as source terms, via removal theory, for the diffusion group fluxes. The removal diffusion transfer cross sections are fundamentally different from those of discrete ordinates since they are based on removal cross sections and define Σ_R in terms of the fundamental cross-section data and an adjustable parameter.

6.4 Monte Carlo method

The analog Monte Carlo method is the most rigorous method of radiation transport analysis. Its underlying assumption is that solutions of the Boltzmann equation can be obtained through random number simulations of the physical transport of particles. The user creates fictitious particles and, using the same probability laws that govern the interactions undergone by the real particle (scattering, leakage, absorption), traces out—step by step and by random choices—a case history to associate with each such particle. A large number of particles are treated. The total set of events contained in the histories of all the quasi particles represents a set of events that might have happened to an equivalent number of real particles if they were put in the same environment described by the problem. Average values com-

puted from a sample size that is sufficiently large and randomly chosen will provide an estimate of the averages determined from actual particle samples.

The average value of the calculated quantity of interest is the ultimate objective. Each estimate of the quantity, which is based on each source particle, is a sample of a random variable, and the entire set of estimates is a sample from a probability distribution function. The variance of the mean, and the square root of the variance (the standard deviation), are measures of the relative spread of this distribution. The smaller the standard deviation is, the closer the average value of a given set of histories is to approximating the mean of the true population. One way to reduce the standard deviation is to increase the number of histories in the sample.

An important concept in the technique is that of statistical weight. All particles originating from the source are assigned a unit weight, unless importance sampling is implemented for specific parameters, e.g., energy or direction. As the particles traverse the media of the geometric model of a given problem, this weight is usually diminished to account for the loss mechanisms. Since all collisions in this scheme are treated as scattering events, the weight after collision of a particle with incident energy E and incident weight W is determined by multiplying W by the ratio of the scattering-to-total interaction probabilities.

Weight is important because it is the quantity that preserves the expected value in some of the variance reduction techniques that are used. These include Russian roulette (in which, based on a selected random number, histories are either continued with an increased weight or terminated), splitting (in which one particle with weight W is split into M particles, each with weight W/M , and each of which subsequently traces out its history independently), and path length stretching (in which the total cross section is altered to extend the paths of particles in preferred directions). Use of these and other techniques, more fully discussed in Refs. [30] and [31], requires proper understanding of their operation. This understanding is important because it is possible to misuse variance reduction methods such that weight is preserved, a small variance in the computed result is

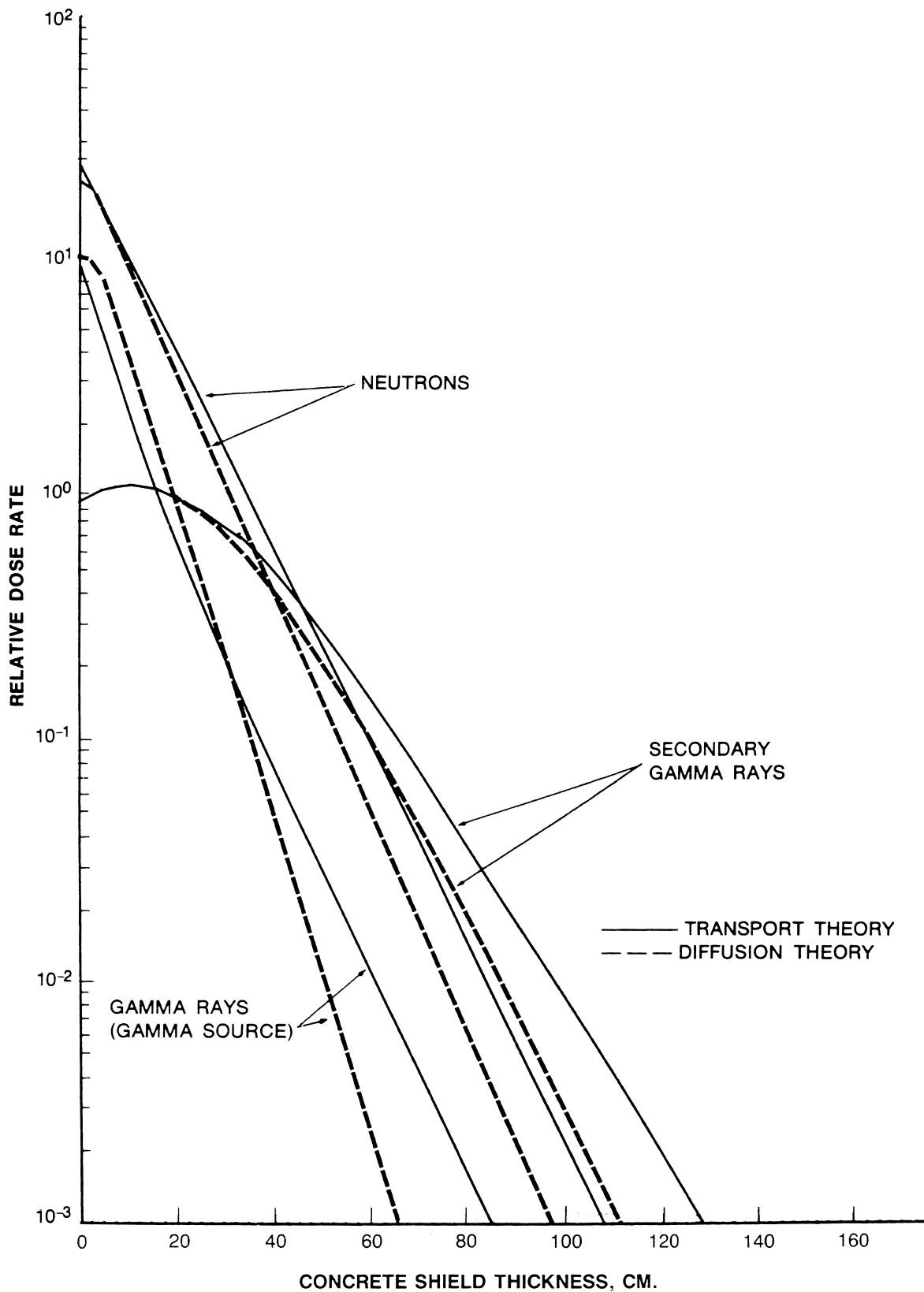


Figure 3 – Comparison of neutron and gamma-ray dose rate through concrete using transport and diffusion theory

achieved, but the computed mean value of the result is incorrect. For further discussion of the potential pitfalls in variance reduction techniques, see Refs. [53] and [54].

In the analog Monte Carlo method, the particles have unit weight, and there is no biasing with accompanying weight changes.

When determining dose rates at various locations, rather than following all histories and then using only those that arrive at the detector locations, the technique of statistical estimation can be used. Each source particle is assumed, through the use of the uncollided point kernel modified by the weight, to contribute to the dose rate. Similarly, at each collision point for each history, the dose contribution is determined, on the assumption that the particle scatters at that point and arrives at the detector, with the weight proportional to the probability of arriving there.

The Monte Carlo method is carried out through the use of computer codes. They allow a very general 3-D geometric description much like the point-kernel codes. Source description is also very general. Specific source points are selected internally on the basis of random numbers. There are two approaches to the energy specification, continuous or multigroup, and this in turn determines the form of the cross-section input data. With a continuous energy format, the scattering cross section can be given by the Klein-Nishina distribution function, but then its use is restricted to gamma rays only. The multigroup format is advantageous because it can be specifically designed to use the same cross-section data as that for the discrete ordinates codes. Where necessary, the multigroup format can be modified to use directly the Klein-Nishina distribution function for the scattering transfer probabilities of gamma rays (see Ref. [55]). Many of the other processes involved utilize a random number generator to select the respective values.

The Monte Carlo method is well suited for scattering problems in complex geometries because it does not use the buildup factor concept (as in point-kernel codes) but rather accounts for the scattered contributions directly. For symmetric configurations where most of the media are dense materials and where the radiation gradient predominates in one or two directions, the discrete ordinates method works well and

may be used. Advances in computer technology, however, have greatly increased the practicality of the Monte Carlo method.

6.5 Other methods and summary

Still other techniques are used in shield analysis: matrix methods, direct integration methods, the moments method, and others. They are generally not very useful for practical shield design. While a shielding engineer may use any calculational method as long as his or her design accommodates the calculational uncertainties, the following observations may be used as a guide in the selection of one of these methods for treatment of a particular problem. The point-kernel method provides the simplest, most straightforward approach when it can be applied validly. The discrete ordinates and Monte Carlo methods are considerably more complex and require greater user experience. The discrete ordinates method provides a detailed map of the radiation field and is best applied where most of the media are dense materials and deep penetration is involved. The Monte Carlo method is particularly useful in the treatment of radiation transport in complex, asymmetric, 3-D configurations where scattering is important; in some cases, however, considerable user experience might be required where variance reduction techniques are required in lieu of running an inordinate number of particle histories.

The following are typical applications of these three principal methods of shield design. The point-kernel method may be used in the design of shielding for equipment that contains gamma-emitting fluids, such as demineralizers, heat exchangers, filters, pipes, tanks, steam lines, etc. The discrete ordinates method is used to design the primary reactor shield because it readily treats coupled neutron and gamma attenuation. The Monte Carlo method is used for complex radiation transport problems that involve scattering, such as neutron streaming or skyshine.

7 Concrete shielding data

7.1 Introduction

The collection of concrete shielding data presented in this section is applicable to most prac-

tical nuclear power plant shielding designs. The major contributions to the body of shielding data collected are from the works of Walker and Grotenhuis [52], Hungerford (Chap. 9 of Ref. [6]), Schmidt [56], Hubbell [57], Hubbell and Seltzer [58], Eisenhauer and Simmons [59], the RSICC, and other references cited in Appendices B and C.

Although there has been a painstaking effort to develop reliable shielding data, the shielding of nuclear power plants remains somewhat less than an exact science. This situation results, in part, from the uncertainty in the properties of the material itself. Concrete is known to vary both in composition and density owing to the wide variety of aggregates. Since the aggregate depends on the origin of the material, the composition of the concrete used for shielding is very dependent upon the plant site.

No credit is taken in the data tables for shielding from large amounts of reinforcing steel interlaced in the concrete to increase its structural strength. From the standpoint of neutron shielding, it is desirable that concrete have a high water content. In practice, since concrete generally serves not only as a shield but also as a structural support of the plant itself, the water content is maintained at a minimum because the structural strength of the concrete decreases as water content increases.

7.2 Gamma-ray attenuation coefficients

Reactor gamma rays are generally classified in two categories: primary gamma rays, from the fission process or the decay of fission products, and secondary gamma rays, from the interaction of neutrons with matter, either by neutron capture or inelastic scattering. Gamma rays from other plant systems arise from radioactive decay of activation or fission products.

When gamma-ray attenuation in shielding design is considered, the gamma rays are usually divided into different energy groups. The number of energy groups depends upon the specific problem and the accuracy desired. Typically, 10 to 20 groups are chosen.

Generally, gamma rays are attenuated in proportion to the atomic mass of the shielding material. The linear attenuation coefficient increases with material density for gamma rays of a given energy group. Since gamma-ray attenuation is dependent upon the shielding material, the composition and density of the material must be known. This requirement presents a problem in concrete shielding designs, owing to the inherent variation in the composition of the concrete. Most of the present literature surveyed references ordinary concrete as having a theoretical density of 2.35 g/cm³. Heavy concrete is included in this standard for reference purposes only.⁸⁾

The values for the mass attenuation coefficient for any mixture $(\mu/\rho)_{mix}$ can be found from

$$(\mu/\rho)_{mix} = \sum_{i=1}^N (\mu/\rho)_i f_{wi} ,$$

where:

$(\mu/\rho)_i$ = mass attenuation coefficient of the i 'th element;

N = total number of elements;

f_{wi} = fractional weight of the i 'th element.⁹⁾

7.3 Gamma-ray buildup factors

Buildup factors are introduced in point-kernel shielding calculations to account for the contribution of scattered photons to the total amount

⁸⁾ Table B.1 lists the values of mass attenuation coefficients for concretes characterized in Table 1. Table B.2 lists the mass attenuation coefficients for the elements that comprise ordinary concrete. Table B.3 lists similar data for additional elements that are found in other concrete compositions.

⁹⁾ The mass energy transfer and energy absorption coefficients for concretes characterized in Table 1 are listed in Tables B.4 and B.5. In estimating absorbed dose in concrete, the former should be used in point-kernel calculations in association with buildup factors. The latter should be used in transport calculations that enumerate fluences of secondary and tertiary photons. The energy absorption coefficient is that part of the attenuation coefficient resulting from energy absorption only and is equal to the product of the energy transfer coefficient and $1 - g$, where g = the fraction of the energy of secondary charged particles that is lost to bremsstrahlung, as well as in-flight annihilation, and other minor photon escape phenomena (see Ref. [58]).

of radiation seen at the point of observation.¹⁰⁾ The buildup factors recommended for use in this standard have been determined from infinite medium moments method calculations. Eisenhauer and Simmons [59] have performed moments method calculations for ordinary concrete, the composition of which is presented in NSRDS-NBS-29 [57]. Unless the concrete in question is substantially different from the concrete used in these calculations, these data should be used rather than those based on an effective atomic number. The cases performed include an isotropic plane source, and empirical functions were developed for dose and energy absorption buildup factors. The results of the relations for point isotropic infinite medium buildup factors are in good agreement with the work of Morris et al. [61], with only small differences, which occur at deep penetrations and at energies of <0.2 MeV. A commonly used fitting function for a point isotropic source is the Taylor formula:

$$B(x) = A \exp(-\alpha_1 x) + (1 - A) \exp(-\alpha_2 x) ,$$

where:

x = thickness in mean free paths.¹¹⁾

Dose (air kerma) and energy absorption (concrete kerma) buildup factors have been based on Ref. [59].¹²⁾ While the fitting function shown above has been satisfactorily used for many applications, it is now known that the best fit to the buildup factor data is that of the geometric progression (G-P) form [4]. The G-P function for the buildup factor $B(E, x)$ is

$$B(E, x) = 1 + (b - 1)(K^x - 1)/(K - 1), \text{ for } K \neq 1$$

and

$$B(E, x) = 1 + (b - 1)x \text{ for } K = 1 ,$$

where:

$$K(x) = cx^a + d[\tanh(x/X_K - 2) - \tanh(-2)]/[1 - \tanh(-2)];$$

a, c, d, X_K = fitting parameters that depend on the source energy E ;

x = distance from source to detector (mfp);

b = buildup factor at 1 mfp.

The G-P form¹³⁾ reproduces the buildup factors to within a few percent for 0 to 40 mfp and for energies of 0.015 to 15.0 MeV and is the recommended form for buildup factors. Included in the tables are deviation data indicating the accuracy of the fit.¹⁴⁾ Implementation of the G-P form provides smooth interpolation in energy. The methodology has been implemented in various point-kernel codes available from RSICC.^{15), 16)}

7.4 Secondary-gamma-ray production

Secondary-gamma-ray production due to thermal neutron capture is important in the design of concrete primary reactor shields for nuclear power plants.¹⁷⁾

7.5 Neutron cross sections

For preliminary calculations involving simple point-kernel removal theory or Spinney's removal-

¹⁰⁾ See Goldstein [60] for early approximations to concrete buildup factors based on those for elements with proximate atomic numbers.

¹¹⁾ Parameters for the Taylor formula are listed in Table B.6 for thicknesses up to 40 mfp.

¹²⁾ Graphs of the buildup factors are presented in Figs. B.1 through B.8 as a function of thickness in mean free paths.

¹³⁾ Coefficients for the G-P form obtained from ANS-6.4.3 [4] are presented in Tables B.7 and B.8 for air kerma in concrete and concrete kerma in concrete buildup factors, respectively.

¹⁴⁾ For example, in Table B.7, the maximum deviation is 2.9% occurring at 0.06 MeV for 0.5 mfp.

¹⁵⁾ See Appendix A.

¹⁶⁾ The point isotropic buildup factors generated by use of the G-P methodology for ordinary concrete are presented in Figs. B.3 through B.8.

¹⁷⁾ Table B.9 lists the resultant gamma-ray spectra from thermal neutron capture in iron, aluminum, silicon, calcium, potassium, and sodium—elements commonly found in concrete.