

# American Nuclear Society

**REAFFIRMED**

**July 23, 2004**  
**ANSI/ANS-6.4-1997 (R2004)**

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

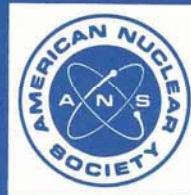
an American National Standard

**WITHDRAWN**

**September 29, 2006**  
**ANSI/ANS-6.4-1997;R2004**

**No longer being maintained as  
an American National Standard.**

**This standard may contain  
outdated material or may have  
been superseded by another  
standard. Please contact the  
ANS Standards Administrator  
for details.**



published by the  
American Nuclear Society  
555 North Kensington Avenue  
La Grange Park, Illinois 60525 USA

**ANSI/ANS-6.4-1997**

**American National Standard  
for 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**

Published by the  
**American Nuclear Society  
555 North Kensington Avenue  
La Grange Park, Illinois 60525 USA**

Approved May 28, 1997  
by the  
**American National Standards Institute, Inc.**

## **American National Standard**

Designation of this document as an American National Standard attests that the principles of openness and due process have been followed in the approval procedure and that a consensus of those directly and materially affected by the standard has been achieved.

This standard was developed under procedures of the Standards Committee of the American Nuclear Society; these procedures are accredited by the American National Standards Institute, Inc., as meeting the criteria for American National Standards. The consensus committee that approved the standard was balanced to ensure that competent, concerned, and varied interests have had an opportunity to participate.

An American National Standard is intended to aid industry, consumers, governmental agencies, and general interest groups. Its use is entirely voluntary. The existence of an American National Standard, in and of itself, does not preclude anyone from manufacturing, marketing, purchasing, or using products, processes, or procedures not conforming to the standard.

By publication of this standard, the American Nuclear Society does not insure anyone utilizing the standard against liability allegedly arising from or after its use. The content of this standard reflects acceptable practice at the time of its approval and publication. Changes, if any, occurring through developments in the state of the art, may be considered at the time that the standard is subjected to periodic review. It may be reaffirmed, revised, or withdrawn at any time in accordance with established procedures. Users of this standard are cautioned to determine the validity of copies in their possession and to establish that they are of the latest issue.

The American Nuclear Society accepts no responsibility for interpretations of this standard made by any individual or by any ad hoc group of individuals. Requests for interpretation should be sent to the Standards Department at Society Headquarters. Action will be taken to provide appropriate response in accordance with established procedures that ensure consensus on the interpretation.

Comments on this standard are encouraged and should be sent to Society Headquarters.

Published by

**American Nuclear Society  
555 North Kensington Avenue  
La Grange Park, Illinois 60526 USA**

Copyright © 1997 by American Nuclear Society. All rights reserved.

Any part of this standard may be quoted. Credit lines should read "Extracted from American National Standard ANSI/ANS-6.4-1997 with permission of the publisher, the American Nuclear Society." Reproduction prohibited under copyright convention unless written permission is granted by the American Nuclear Society.

Printed in the United States of America

## Foreword

(This Foreword is not a part of American National Standard for Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, ANSI/ANS-6.4-1997.)

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, there have been a number of other advances, particularly with respect to buildup factors. These advances have prompted this newest revision, ANSI/ANS-6.4-1997.

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:

J. L. Kamphouse, Chairman, *Tennessee Valley Authority*  
R. J. Donahue, *Lawrence Berkeley National Laboratory*  
S. J. Haynes, *Washington Public Power Supply System*  
A. R. Larson, *Bechtel Corporation*  
R. W. Roussin, *Oak Ridge National Laboratory*  
J. K. Warkentin, *TU Electric Company*

Subcommittee ANS-6, Radiation Protection and Shielding, had the following membership at the time of its approval of this standard:

W. C. Hopkins, Chairman, *Bechtel Corporation*  
D. R. Harris, Jr., *Rensselaer Polytechnic Institute*  
J. L. Kamphouse, *Tennessee Valley Authority*  
D. C. Kaul, *Science Applications International Corporation*  
R. T. Klann, *Argonne National Laboratory-West*  
D. K. Trubey, *Individual*  
N. Tsoulfanidis, *University of Missouri-Rolla*

Consensus Committee N-17, Research Reactors, Reactor Physics, Radiation Shielding, and Computational Methods, had the following membership at the time it reviewed and approved this standard:

T. M. Raby, Chairman  
A. Weitzberg, Secretary

H. Alter	U. S. Department of Energy
A. D. Callihan	Individual
R. E. Carter	Individual
D. Cokinos	Brookhaven National Laboratory
A. De La Paz	Vista Technologies
B. Dodd	Health Physics Society
D. Duffey	American Institute of Chemical Engineers
W. A. Holt	American Public Health Association
W. C. Hopkins	Bechtel Corporation
L. I. Kopp	Individual
J. Miller	Institute of Electrical and Electronics Engineers, Inc.
J. E. Olhoeft	Individual
T. M. Raby	American Nuclear Society
W. J. Richards	U. S. Department of Defense
T. R. Schmidt	Sandia National Laboratories
R. L. Seale	University of Arizona
A. O. Smetana	Westinghouse Savannah River Company
M. M. Ter Pogossian	American College of Radiology
J. F. Torrence	National Institute of Standards and Technology
D. K. Trubey	Individual
S. H. Weiss	U. S. Nuclear Regulatory Commission
A. Adams Jr. (Alt.)	
A. Weitzberg	Halliburton NUS Corporation
W. L. Whittemore	GA Technologies, Inc.

<b>Contents</b>	<b>Section</b>	<b>Page</b>
1. Scope .....		1
2. Requirements and Recommendations .....		1
2.1 Conformance .....		1
2.2 Requirements .....		1
2.3 Recommendations .....		2
3. Standards of Documentation .....		2
3.1 Shield Design Approach .....		3
3.2 Shield Design Description .....		3
3.3 Methods of Analysis .....		3
3.4 Description of Analyses .....		3
3.5 References .....		3
3.6 Summary Results .....		3
4. Terms and Definitions .....		3
5. Characterization of Concrete .....		5
5.1 Introduction .....		5
5.2 Concrete Placement .....		6
5.3 Water Content .....		6
5.4 Heating Effects .....		7
5.5 Reinforcing Steel .....		7
5.6 Aggregates .....		7
6. Calculation Methods .....		10
6.1 Introduction .....		10
6.2 Point Kernel Methods .....		10
6.3 Discrete Ordinates Method .....		12
6.4 Monte Carlo Method .....		16
6.5 Other Methods, and Summary .....		18
7. Concrete Shielding Data .....		19
7.1 Introduction .....		19
7.2 Gamma Ray Attenuation Coefficients .....		19
7.3 Gamma Ray Buildup Factors .....		20
7.4 Secondary Gamma Ray Production .....		21
7.5 Neutron Cross Sections .....		21
7.6 Neutron Attenuation Curves .....		21
8. Applications .....		21
8.1 Radiation Effects .....		21
8.2 Minimum Water Content .....		22
8.3 Bulk Transport .....		23
8.4 Radiation Streaming Through Penetrations .....		25
8.5 Reflection .....		26
9. References .....		27
Bibliography .....		33

Appendices

Appendix A	List of Codes .....	34
Appendix B	Shielding Data .....	36
Appendix C	Applications Data and Results .....	60

Tables

Table 5.1	Partial Densities of Concrete Constituents .....	5
Table 5.2	Typical Concrete Properties .....	9
Table B.1	Mass Attenuation Coefficients (cm <sup>2</sup> /g) for Ordinary Concrete .....	36
Table B.2	Mass Attenuation Coefficients for Elements Comprising Ordinary Concrete .....	37-38
Table B.3	Mass Attenuation Coefficients of Elements Which May Be Found in Other Concrete Compositions (cm <sup>2</sup> /g) .....	39
Table B.4	Mass Energy Absorption Coefficients for Ordinary Concrete (cm <sup>2</sup> /g) .....	40
Table B.5	Gamma Ray Energy Absorption Coefficients of Various Types of Concrete (cm <sup>-1</sup> ) .....	40
Table B.6	The Equivalent Atomic Number for Ordinary Concrete as a Function of Source Energy .....	40
Table B.7	Point Isotropic Infinite Medium Buildup Factor Parameters for Ordinary Concrete (Dose, 20 mfp) .....	41
Table B.8	Point Isotropic Infinite Medium Buildup Factor Parameters for Ordinary Concrete (Energy Absorption, 20 mfp) .....	42
Table B.9	Point Isotropic Infinite Medium Buildup Factor Parameters for Ordinary Concrete (Dose, 40 mfp) .....	43
Table B.10	Point Isotropic Infinite Medium Buildup Factor Parameters for Ordinary Concrete (Energy Absorption, 40 mfp) .....	44
Table B.11	Geometric Progression Buildup Factor Coefficients for Ordinary Concrete (Dose, 40 mfp) .....	45
Table B.12	Geometric Progression Buildup Factor Coefficients for Ordinary Concrete (Energy Absorption, 40 mfp) .....	46
Table B.13	Gamma Ray Spectra from Thermal Neutron Capture in Concrete .....	47
Table B.14	Neutron Constants for Concrete .....	47
Table C.1	Composition of Hanford Ordinary Concrete as a Function of Temperature .....	60
Table C.2	Compositions of Concretes Used in Measurements .....	60
Table C.3	ANISN Spherical Model for PWR Calculation .....	61
Table C.4	Material Compositions for PWR Calculation .....	61
Table C.5	Neutron Source Distribution for PWR Calculation .....	62
Table C.6	Neutron Source Spectrum for PWR Calculation .....	62
Table C.7	Albedo Method Parameters .....	63
Table C.8	Constants for the Expression Fitting the Maerker- Muckenthaler Differential Dose Albedo Data for Fast Neutrons Incident on Concrete .....	64
Table C.9	Value of Parameters for Chilton-Huddleston Gamma Ray Differential Albedo Formula .....	64
Table C.10	Constants for the Expressions Fitting the Coleman <i>et al.</i> Differential and Total Albedo Data for Intermediate- Energy Neutrons Incident on Reinforced Concrete .....	65

Figures

Figure 6.1	Example of Oblique Penetration Short Circuit Paths	12
Figure 6.2	$P_1$ Convergence of the Fast Neutron Dose Rate in Water from a Point Fission Source	15
Figure 6.3	Comparison of Neutron and Gamma Ray Dose Rate Through Concrete Using Transport and Diffusion Theory	17
Figure 8.1	Example of Excess Transmission Through a Void	25
Figure B.1	Dose Buildup Factor, Point Isotropic Source	48
Figure B.2	Dose Buildup Factor Geometric Progression Form (Energy: 0.03-0.2 MeV)	49
Figure B.3	Dose Buildup Factor Geometric Progression Form (Energy: 0.3-2.0 MeV)	50
Figure B.4	Dose Buildup Factor Geometric Progression Form (Energy: 3.0-15.0 MeV)	51
Figure B.5	Dose Buildup Factor, Plane Isotropic Source	52
Figure B.6	Energy Absorption Buildup Factor, Point Source	53
Figure B.7	Energy Absorption Buildup Factor Geometric Progression Form (Energy: 0.03-0.2 MeV)	54
Figure B.8	Energy Absorption Buildup Factor Geometric Progression Form (Energy: 0.3-2.0 MeV)	55
Figure B.9	Energy Absorption Buildup Factor Geometric Progression Form (Energy: 3.0-15.0 MeV)	56
Figure B.10	Energy Absorption Buildup Factor, Plane Source	57
Figure B.11	Comparison of Adjoint Discrete Ordinates and Moments Method Calculations of Neutron Dose Equivalent (rems) from a Plane Isotropic Fission Source as a Function of Concrete Thickness, Source Normalized to One Neutron in a Forward Direction	58
Figure C.1	Resonance Neutron Distribution in Ordinary Concrete as a Function of Temperature	66
Figure C.2	Fast Neutron Distribution in Ordinary Concrete as a Function of Temperature	67
Figure C.3	Measured Neutron Fluxes vs. Distance in Ordinary Concrete	68
Figure C.4	Gamma Ray and Neutron Flux Attenuation Curves in Ordinary Concrete	69
Figure C.5	Radiation Levels in Bradwell Side Shield	70
Figure C.6	Dose Rates in Typical PWR Shield	71
Figure C.7	Gamma Dose Rates from 4-Inch Schedule 160 Pipe Containing Radioactivity, Derived Through the Use of the PATH Code	72
Figure C.8	Gamma Dose Rates from 24-Inch Schedule 160 Pipe Containing Radioactivity, Derived Through the Use of the PATH Code	73
Figure C.9	Gamma Dose Rates from 6-Foot Outer Diameter Tank Containing Radioactivity, Derived Through the Use of the PATH Code	74
Figure C.10	Penetration Types	75
Figure C.11	Geometry for Albedos	76
Figure C.12	Reflection Geometries	77



# 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 which 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" to denote a recommendation; and the word "may" 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 which are exposed to an incident energy flux greater than  $10^{10}$  MeV/cm<sup>2</sup>-sec and which 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 8.4 of this standard.
- (2) Determination that the configuration is similar to one which is functioning properly under comparable conditions in an operating nuclear facility.
- (3) Determination that the configuration is similar to one which has been evaluated experimentally and found to be effective for the radiation levels under consideration.
- (4) Determination that the configuration is similar to one which has already been shown by analysis to be effective.