

# American Nuclear Society

**WITHDRAWN**

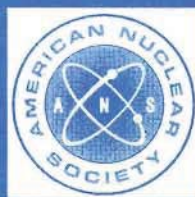
May 19, 2000

ANSI/ANS-19.1-1983;R1988

**nuclear safety criteria for the design of  
stationary pressurized water reactor plants**

**an American National Standard**

**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standard may contain outdated material  
or may have been superseded by another  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 60526 USA**

**American National Standard  
Nuclear Safety Criteria for the Design of  
Stationary Pressurized Water Reactor Plants**

**Secretariat  
American Nuclear Society**

**Prepared by the  
American Nuclear Society  
Standards Committee  
Working Group ANS-51.1**

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

**Approved April 29, 1983; Reaffirmed September 26, 1988  
by the  
American National Standards Institute, Inc.**

## **American National Standard**

An American National Standard implies a consensus of those substantially concerned with its scope and provisions. An American National Standard is intended as a guide to aid the manufacturer, the consumer, and the general public. The existence of an American National Standard does not in any respect preclude anyone, whether he has approved the standard or not, from manufacturing, marketing, purchasing, or using products, processes, or procedures not conforming to the standard. American National Standards are subject to periodic review and users are cautioned to obtain the latest editions.

**CAUTION NOTICE:** This American National Standard may be reviewed or withdrawn at any time. The procedures of the American National Standards Institute require that action be taken to reaffirm, revise, or withdraw this standard no later than five years from the date of publication. Purchasers of this standard may receive current information, including interpretation, on all standards published by the American Nuclear Society by calling or writing to the Society.

Published by

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

Copyright © 1983 by American Nuclear Society.

Any part of this standard may be quoted. Credit lines should read "Extracted from American National Standard ANSI/ANS-51.1-1983(R1988) 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 Standards Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANSI/ANS-51.1-1983(R1988).)

This standard is a complete revision and combination of N18.2-1973/ANS-51.1 and N18.2a-1975/ANS-51.8. It has been prepared by Subcommittee ANS-51, Pressurized Water Reactor Criteria, to incorporate additional requirements for the design of pressurized water reactor (PWR) nuclear power plants and to address three major areas:

## 1. Safety Classes

The results of the ANS Nuclear Power Plant Standards Committee (NUPPSCO) Ad Hoc Committee and NUPPSCO Coordinating Working Group 3 on Equipment Classification are incorporated. These results define Safety Classes and specify requirements for all equipment and structures in a stationary nuclear power plant having a nuclear safety function. A methodology is given to classify all equipment into one of three Safety Classes according to its importance to nuclear safety and its capability for maintenance, surveillance testing, and inspection, or into a Non-Nuclear Safety Class. In addition, classification interface criteria are defined.

## 2. Plant Conditions

The results of the NUPPSCO Coordinating Working Group 2 have been incorporated. The concept of Plant Conditions is developed that includes individual process conditions, combinations of process conditions, and the combinations of process conditions and external hazards that could result in simultaneous effects on plant equipment. Probability of occurrence is the unifying basis for the categorization of Plant Conditions.

## 3. Design Requirements

This standard provides a set of design requirements for all Safety Classes and Non-Nuclear Safety Class in terms of industry codes and standards for each category of Plant Conditions. The design requirements reference specific standards and ensure substantial interrelationship with other codes and standards.

The content of this standard reflects an attempt to achieve the following objectives:

- a. To establish a consistent set of requirements for light water reactor nuclear power plants;
- b. To establish a disciplined, systematic method for defining nuclear safety requirements for nuclear power plants;
- c. To establish and delineate the functional nuclear safety requirements for the design of nuclear power plants;
- d. To be responsive to both the regulatory requirements of the Nuclear Regulatory Commission and the design and technical requirements of industry codes and standards;
- e. To provide a framework for augmenting these criteria as additional standards are developed within the nuclear industry; and
- f. To provide a uniform basis for design safety requirements which may be reflected in regulatory documents.

The existence of unique plant or site characteristics might require the consideration of alternate design concepts. This standard has been developed along functional lines to permit this flexibility. The standard has, however, cited many standards, some of which were still in draft form at the time this document was published. Provisions



contained in any draft standard should be considered and used with great discretion. It is strongly suggested that the prospective user fully understand the present status of the referenced standard and major factors on why it might be still in draft form; for example, controversial issues should be recognized.

A number of considerations under development concurrent with the preparation of this standard are not addressed in this standard. Examples of these considerations include: human factors engineering (HFE), probabilistic risk assessment (PRA), systems interaction, diversity, plant security, emergency response facilities, degraded core, minimizing challenges to engineered safety features, safety goals and consideration of cost/benefit analysis, and anticipated transients without scram. Subsequent revisions of this standard will address these considerations as appropriate when they become adequately defined.

A designer is not restricted by this standard from proposing or using alternate criteria to ensure adequate nuclear safety. Frequently, a desirable overall result can be obtained by any of several design concepts. The designer may choose from several alternatives in satisfying the specifics of this standard by the proper consideration of the interrelationship of components and systems within the plant. For example, the PRA approach may be used as an alternative method to evaluate plant design; however, its usefulness is somewhat limited without safety goals that are currently under development.

Portions of this standard were prepared separately under ANS-50 Nuclear Power Plant Systems Engineering and were reviewed individually by ANS-51, ANS-50, and NUPPSCO which replaced ANS-50 during this time. The separate documents that have been incorporated into this standard include the Glossary (CWG-1), Conditions of Design (CWG-2), and Equipment Classification (Ad Hoc Committee on Equipment Classification). The structure of this standard is based on the standard format guide (CWG-4). This standard was approved by NUPPSCO in 1982.

This standard and all other ANS standards have been written for prospective use.

Continuing efforts will be required to augment or modify the criteria in this standard to implement changing licensing requirements, to achieve standardization among the various industry criteria and standards currently being developed, and to provide additional clarification or interpretation as appropriate. The ANS-51 PWR Criteria Committee meets periodically to consider revisions or modifications to this standard.

Comments, suggestions, and requests for interpretations should be addressed to the Chairman, ANS-51 PWR Criteria Committee, American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Ill. 60525.

Working Group ANS-51.1 of the Standards Committee of the American Nuclear Society, had the following membership at the time it developed this standard:

|                                                                     |                                                        |
|---------------------------------------------------------------------|--------------------------------------------------------|
| G. A. Zimmerman, Chairman, <i>Portland General Electric Company</i> | R. C. Surman, <i>Westinghouse Electric Corporation</i> |
|---------------------------------------------------------------------|--------------------------------------------------------|

At the time of its approval of this standard, Subcommittee ANS-51 had the following membership:

|                                                             |                                                           |
|-------------------------------------------------------------|-----------------------------------------------------------|
| C. J. Gill, Chairman, <i>Bechtel Power Corporation</i>      | W. Moody, <i>Southern California Edison</i>               |
| R. Bone, <i>Stone &amp; Webster Engineering Corporation</i> | H. G. O'Brien, <i>Tennessee Valley Authority</i>          |
| V. M. Callaghan, <i>Combustion Engineering, Inc.</i>        | R. Shone, <i>Yankee Atomic Electric Company</i>           |
| R. W. Fleming, <i>Westinghouse Electric Corporation</i>     | E. W. Swanson, <i>Babcock &amp; Wilcox Company</i>        |
| M. Horrell, <i>Ebasco Services, Inc.</i>                    | G. A. Zimmerman, <i>Portland General Electric Company</i> |
| D. Lewis, <i>Bechtel Power Corporation</i>                  |                                                           |

The American Nuclear Society's Nuclear Power Plant Standards Committee (NUPPSCO) had the following membership at the time of its approval of this standard.

L. J. Cooper, Chairman  
M. D. Weber, Secretary

| Name of Representative | Organization Represented                                                                    |
|------------------------|---------------------------------------------------------------------------------------------|
| R. G. Benham           | General Atomic Company<br>(for the Institute of Electrical and Electronics Engineers, Inc.) |
| B. M. Rice (Alt.)      | Duke Power Company<br>(for the Institute of Electrical and Electronics Engineers, Inc.)     |
| R. V. Bettinger        | Pacific Gas and Electric Company                                                            |
| P. Bradbury            | Westinghouse Advanced Reactor Division                                                      |
| D. A. Campbell         | Westinghouse Electric Corporation                                                           |
| C. O. Coffey           | Pacific Gas and Electric Company                                                            |
| L. J. Cooper           | Nebraska Public Power District<br>(for the American Nuclear Society)                        |
| W. H. D'Ardenne        | General Electric Company                                                                    |
| S. B. Gerges           | NUS Corporation                                                                             |
| C. J. Gill             | Bechtel National, Inc.                                                                      |
| C. Johnson             | U.S. Nuclear Regulatory Commission                                                          |
| R. W. Keaten           | GPU Services Corporation                                                                    |
| J. W. Lentsch          | Portland General Electric Company                                                           |
| J. F. Mallay           | NUTECH Engineers*                                                                           |
| L. M. Mills            | Tennessee Valley Authority                                                                  |
| A. T. Molin            | United Engineers and Constructors, Inc.                                                     |
| T. J. Pashos           | Individual                                                                                  |
| P.T. Reichert          | Catalytic, Inc.                                                                             |
| S. L. Rosen            | Institute of Nuclear Power Operations                                                       |
| J. E. Smith            | Duke Power Company                                                                          |
| J. W. Stacey           | Yankee Atomic Electric Company                                                              |
| S. L. Stamm            | Stone & Webster Engineering Corporation                                                     |
| L. Stanley             | Quadrex/Nuclear Services Corporation                                                        |
| J. D. Stevenson        | Structural Mechanics Associates<br>(for the American Society of Civil Engineers)            |
| C. D. Thomas           | Science Applications, Inc.                                                                  |
| G. P. Wagner           | Commonwealth Edison Company                                                                 |
| G. L. Wessman          | Torrey Pines Technology                                                                     |
| J. E. Windhorst        | Individual                                                                                  |

\*Formerly with Babcock & Wilcox Company

| <b>Contents</b>                                                                                    | <b>Section</b> | <b>Page</b> |
|----------------------------------------------------------------------------------------------------|----------------|-------------|
| 1. Introduction . . . . .                                                                          | 1              | 1           |
| 1.1 Scope . . . . .                                                                                | 1              | 1           |
| 1.2 Purpose . . . . .                                                                              | 1              | 1           |
| 2. Definitions . . . . .                                                                           | 2              | 2           |
| 3. General Safety Criteria . . . . .                                                               | 5              | 5           |
| 3.1 General Approach . . . . .                                                                     | 5              | 5           |
| 3.2 Plant Conditions and Plant Nuclear Safety Criteria . . . . .                                   | 6              | 6           |
| 3.2.1 Application of the Single Failure Criterion . . . . .                                        | 7              | 7           |
| 3.2.2 Coincident Occurrences . . . . .                                                             | 8              | 8           |
| 3.2.3 Optional Approach . . . . .                                                                  | 9              | 9           |
| 3.2.4 Plant Condition Application Examples . . . . .                                               | 9              | 9           |
| 3.2.5 Multiple Failures in Nuclear Safety-Related<br>Equipment and Common Cause Failures . . . . . | 10             | 10          |
| 3.2.6 Operator Action and Human Error . . . . .                                                    | 10             | 10          |
| 3.2.7 Site Conditions . . . . .                                                                    | 10             | 10          |
| 3.2.8 Natural and Man-Made Hazards . . . . .                                                       | 10             | 10          |
| 3.3 Equipment Classification . . . . .                                                             | 11             | 11          |
| 3.3.1 Safety Classes . . . . .                                                                     | 11             | 11          |
| 3.3.2 Safety Class Interfaces . . . . .                                                            | 13             | 13          |
| 3.3.3 Safety Class Requirements Correlations . . . . .                                             | 16             | 16          |
| 3.4 Industry Codes and Standards . . . . .                                                         | 16             | 16          |
| 3.4.1 Safety Class 1, 2, and 3 Mechanical Equipment . . . . .                                      | 16             | 16          |
| 3.4.2 Safety Class 3 Electrical Equipment . . . . .                                                | 17             | 17          |
| 3.4.3 Safety Class 2 and 3 Structures . . . . .                                                    | 17             | 17          |
| 3.4.4 Non-Nuclear Safety Equipment . . . . .                                                       | 17             | 17          |
| 3.4.5 Quality Assurance . . . . .                                                                  | 17             | 17          |
| 3.5 Safety Analyses . . . . .                                                                      | 18             | 18          |
| 3.5.1 General Requirements . . . . .                                                               | 18             | 18          |
| 3.5.2 Requirements for Plant Condition 1 . . . . .                                                 | 18             | 18          |
| 3.5.3 Requirements for Other Plant Conditions . . . . .                                            | 18             | 18          |
| 4. Design Criteria* . . . . .                                                                      | 32             | 32          |
| 4.1 Reactor Core and Internals . . . . .                                                           | 32             | 32          |
| 4.2 Reactivity Control Systems . . . . .                                                           | 33             | 33          |
| 4.3 Protection System . . . . .                                                                    | 34             | 34          |
| 4.4 Reactor Coolant System . . . . .                                                               | 34             | 34          |
| 4.5 Shutdown Heat Removal Systems . . . . .                                                        | 36             | 36          |
| 4.6 Reactor Coolant Auxiliary Systems . . . . .                                                    | 37             | 37          |
| 4.7 Cooling Water Systems . . . . .                                                                | 38             | 38          |
| 4.8 Emergency Core Cooling Systems . . . . .                                                       | 39             | 39          |
| 4.9 Primary Containment . . . . .                                                                  | 40             | 40          |
| 4.10 Emergency Secondary Heat Removal Systems . . . . .                                            | 41             | 41          |
| 4.11 Containment Auxiliary Systems . . . . .                                                       | 43             | 43          |
| 4.12 Safety-Related Area Cooling Systems . . . . .                                                 | 43             | 43          |
| 4.13 Fuel Storage and Handling . . . . .                                                           | 44             | 44          |
| 4.14 Electrical Power Systems . . . . .                                                            | 44             | 44          |
| 4.15 Fire Protection Systems . . . . .                                                             | 45             | 45          |
| 4.16 Control Complex . . . . .                                                                     | 45             | 45          |

|      |                                      |    |
|------|--------------------------------------|----|
| 4.17 | Radioactive Waste Processing Systems | 46 |
| 4.18 | Other Structures                     | 47 |
| 4.19 | Power Conversion System              | 47 |
| 4.20 | Multi-Unit Stations                  | 48 |

\*Each subsection of Section 4 adheres to the following outline:

|            |                                                                                                                                  |    |
|------------|----------------------------------------------------------------------------------------------------------------------------------|----|
| 4.X        | Title                                                                                                                            |    |
| 4.X.1      | Function                                                                                                                         |    |
| 4.X.2      | Definition                                                                                                                       |    |
| 4.X.3      | Performance Criteria                                                                                                             |    |
| 4.X.4      | Safety Class                                                                                                                     |    |
| 4.X.5      | Design Criteria                                                                                                                  |    |
| 4.X.5.1    | Nuclear Design Criteria                                                                                                          |    |
| 4.X.5.2    | Systems Design Criteria                                                                                                          |    |
| 4.X.5.3    | Mechanical Design Criteria                                                                                                       |    |
| 4.X.5.4    | Electrical Design Criteria                                                                                                       |    |
| 4.X.5.5    | Instrumentation and Control Design Criteria                                                                                      |    |
| 4.X.5.6    | Structural Design Criteria                                                                                                       |    |
| 4.X.5.7    | Testing and Inspection Criteria                                                                                                  |    |
| 4.X.5.8    | Layout Criteria                                                                                                                  |    |
| 5.         | References                                                                                                                       | 51 |
| Appendices |                                                                                                                                  |    |
| Appendix A | Classification Application                                                                                                       | 55 |
| Appendix B | Basis for Plant Condition Criteria in Subsection 3.2                                                                             | 77 |
| Appendix C | Historical Background and Rationale for Equipment<br>Classification in Subsection 3.3                                            | 86 |
| Appendix D | Basis for Mechanical Design Criteria in Subsection 3.4.1                                                                         | 91 |
| Tables     |                                                                                                                                  |    |
| Table 3-1  | Offsite Radiological Dose Criteria for Plant Conditions                                                                          | 20 |
| Table 3-2  | Plant Nuclear Safety Criteria                                                                                                    | 21 |
| Table 3-3  | Example of a Set of Limiting Normal Operations and<br>Limiting Events                                                            | 22 |
| Table 3-4  | Methodology for Determining the Plant Condition of an Event                                                                      | 24 |
| Table 3-5  | Basic Requirements for Equipment by Safety Class                                                                                 | 25 |
| Table 3-6  | ASME Boiler and Pressure Vessel Code, Section III<br>Service Limits for Various Plant Conditions and<br>Nuclear Safety Functions | 26 |
| Table 3-7  | Standards for Safety Class 3 Electrical Equipment (Class 1E)                                                                     | 27 |
| Table 3-8  | Codes and Standards for Safety Class 2 and 3 Structures                                                                          | 28 |
| Table 4-1  | System Functional Design Criteria Interfaces                                                                                     | 50 |



|                                          |    |
|------------------------------------------|----|
| Table A-1 Equipment Classification ..... | 56 |
|------------------------------------------|----|

|                                                                                                           |    |
|-----------------------------------------------------------------------------------------------------------|----|
| Table A-2 Examples of Typical Classification of Components<br>Comprising Complex Principal Equipment..... | 71 |
|-----------------------------------------------------------------------------------------------------------|----|

#### Figures

|                                                     |    |
|-----------------------------------------------------|----|
| Fig. 3-1 Fluid-System Safety Class Interfaces ..... | 29 |
|-----------------------------------------------------|----|

|                                     |    |
|-------------------------------------|----|
| Fig. B-1 Event Categorization ..... | 83 |
|-------------------------------------|----|

|                                                                     |    |
|---------------------------------------------------------------------|----|
| Fig. B-2 Dose Limit Line for Whole-Body Dose at Site Boundary ..... | 84 |
|---------------------------------------------------------------------|----|

|                                                                  |    |
|------------------------------------------------------------------|----|
| Fig. B-3 Dose Limit Line for Thyroid Dose at Site Boundary ..... | 84 |
|------------------------------------------------------------------|----|

|                                                                                      |    |
|--------------------------------------------------------------------------------------|----|
| Fig. B-4 Whole-Body Dose Limit Line Based on 10 CFR 50<br>Appendix I Guideline ..... | 85 |
|--------------------------------------------------------------------------------------|----|

|                                                                                   |    |
|-----------------------------------------------------------------------------------|----|
| Fig. B-5 Thyroid Dose Limit Line Based on 10 CFR 50<br>Appendix I Guideline ..... | 85 |
|-----------------------------------------------------------------------------------|----|