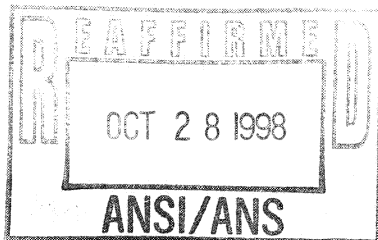


American Nuclear Society

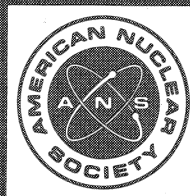
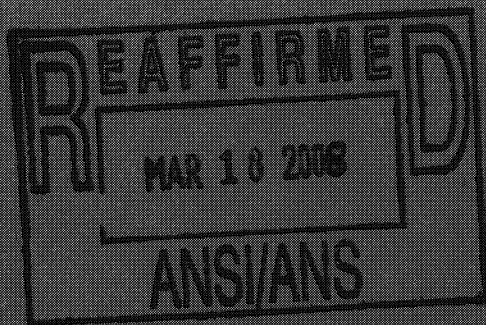


**physical protection for nuclear
safety-related systems
and components**

an American National Standard

This standard has been reviewed and reaffirmed by the ANS Nuclear Facilities Standards Committee (NFSC) with the recognition that it may reference other standards and documents that may have been superceded or withdrawn. The requirements of this document will be met by using the version of the standards and documents referenced herein. It is the responsibility of the user to review each of the references and to determine whether the use of the original references or more recent versions is appropriate for the facility. Variations from the standards and documents referenced in this standard should be evaluated and documented.

This standard does not necessarily reflect recent industry initiatives for risk informed decision-making or a graded approach to quality assurance. Users should consider the use of these industry initiatives in the application of this standard.



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

Addendum to ANSI/ANS-58.3-1992 (R2008) Foreword

[The foreword and this addendum are not part of American National Standard "Physical Protection for Nuclear Safety-Related System and Components," ANSI/ANS-58.3-1992 (R2007)]

The scope of the ANS-58.3 standard is to identify potential natural phenomena and accident safety hazards to light water nuclear power plants and to discuss in general terms means for protecting against such hazards, except for missiles where simplified first order methods to resist missile effects are presented in Appendix B. The term "physical protection" originally was related to accidental and natural hazards. In recent years, it has come to be associated with safeguard requirements associated with acts of sabotage and land, sea, or air malevolent vehicle intrusion.

With the present-day definition of "physical protection," it would be fitting to revise the title to "Natural Phenomena and Accident Hazards for Nuclear Safety-Related Systems and Components." However, American National Standards Institute requirements do not permit any changes to a current standard through reaffirmation. As the foreword is not considered part of the standard, it is acceptable to include this addendum.

The safety hazards identified in this standard are still applicable to the design of nuclear facilities and as such should still be identified in an American Nuclear Society standard. To the extent the standard serves as a guide to the natural phenomena and accident hazards to be considered in nuclear facility design, it serves a useful purpose to the nuclear industry. For this reason the standard was reaffirmed in 2008. Currently, safeguards design requirements that have been identified by regulatory authorities are outside the scope of this standard.

Furthermore, we'd like to acknowledge a large effort by the U.S. Army Corps of Engineers (COE) and the American Society of Civil Engineers (ASCE) on the subject of this standard. It is believed that the COE and ASCE documents will in time cover most of the details in ANS-58.3. Once these documents are finalized, they will be reviewed and addressed in a revision of this standard if determined appropriate.

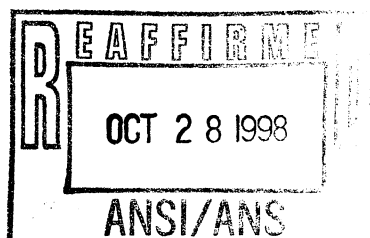
**American National Standard
for Physical Protection for Nuclear
Safety-Related Systems and Components**

Secretariat
American Nuclear Society

Prepared by the
**American Nuclear Society
Standards Committee
Working Group ANS-58.3**

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

Approved August 6, 1992
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 the 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 60525 USA

Copyright © 1993 by American Nuclear Society.

Any part of this standard may be quoted. Credit lines should read "Extracted from American National Standard ANSI/ANS-58.3-1992 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 part of American National Standard Physical Protection for Nuclear Safety-Related Systems and Components, ANSI/ANS-58.3-1992.)

The working group revised this standard based on the assumptions that a stationary light water reactor power plant design team would use this standard in conjunction with many other standards, codes, and regulations. The effort was to make this standard as broad as possible to cause the designers to consider every relevant area based on industry experience to date. Details in any particular area would come from standards specific to that area.

A draft standard on plant design against missiles was issued for trial use and comment in 1974 as ANSI/N177 (ANS-58.1). This work is included as a nonmandatory appendix.

Working Group ANS-58.3 consists of the following members:

H. C. Shaffer III, Chairman, *Yankee Atomic Electric Company*
G. H. Marcus, *U.S. Nuclear Regulatory Commission*
J. Conant, *Combustion Engineering, Inc.*
C. Zeamer, *Washington Public Power Supply System*
R. Harris, *Consultant*

R. C. Surman, *Westinghouse Electric Corporation*, provided assistance in the resolution of probabilistic approaches that are now discussed only in parent standards ANSI/ANS-51.1-1983 (R1988) and ANSI/ANS-52.1-1983 (R1988).

The membership of MC-1, Light Water Reactor Criteria Management, at the time it reviewed this standard was as follows:

W. H. D'Ardenne, Chairman, *GE Nuclear Energy*
R. Bone, *Stone & Webster Engineering Corporation*
E. J. Borella, *Ebasco Services, Inc.*
D. M. Crowe, *Georgia Power & Light Company*
R. Fortier, *Stone & Webster Engineering Corporation*
E. E. Fricks, *Stone & Webster Engineering Corporation*
J. C. Glynn, *U.S. Nuclear Regulatory Commission*
P. Hepner, *Combustion Engineering, Inc.*
D. G. Keith, *Bechtel Power Corporation*
R. J. Mazza, *Sargent & Lundy*
W. Moody, *Southern California Edison*
H. G. O'Brien, *Tennessee Valley Authority*
P. M. Pivawer, *Power Authority of the State of New York*
H. A. Putre, *Perry Nuclear Power Plant*
H. C. Shaffer III, *Yankee Atomic Electric Company*
R. C. Surman, *Westinghouse Electric Corporation*
E. W. Swanson, *Babcock & Wilcox Company*
E. R. Wiot, *Roy F. Weston Company*

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

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

W. M. Andrews	Southern Company Services, Inc.
F. Boorboor	United Engineers & Constructors*
C. O. Coffey	Pacific Gas & Electric Company
L. J. Cooper	Nebraska Public Power District (for the American Nuclear Society)
J. D. Crawford	Combustion Engineering Corporation
W. H. D'Ardenne (Vice Chairman)	GE Nuclear Energy
S. N. Ehrenpreis	Westinghouse Electric Corporation
S. B. Gerges	NUS Corporation, Inc.
C. J. Gill	Bechtel National, Inc.
C. E. Johnson, Jr.	U.S. Nuclear Regulatory Commission
D. Lambert	Tennessee Valley Authority
R. T. Lance	Rockwell International Corporation
J. F. Mallay	Management Analysis Company*
J. K. McCall	Pacific Gas & Electric Company
R. E. Miller	Duke Power Company
J. A. Nevshemal	Toledo Edison Company
P. T. Reichert	United Engineers & Constructors
B. M. Rice	Duke Power Company (retired) (for the Institute of Electrical & Electronics Engineers, Inc.)
J. C. Saldarini	Ebasco Services, Inc.
M. O. Sanford	GPU Nuclear Corporation
S. L. Stamm	Stone & Webster Engineering Corporation
J. D. Stevenson	Stevenson & Associates
T. J. Sullivan	Institute of Nuclear Power Operations
C. D. Thomas, Jr.	Yankee Atomic Electric Company
W. T. Ullrich	Philadelphia Electric Company
G. P. Wagner	Commonwealth Edison Company
G. L. Wessman	Consultant
G. J. Wrobel	Rochester Gas & Electric Corporation

* Affiliation at time of balloting

Contents	Section	Page
1. Introduction and Scope	1	1
1.1 Scope	1	1
1.2 Purpose	1	1
1.3 Limits of Application of the Criteria	1	1
2. Definitions	2	2
3. Design Approach for Providing Protection	2	2
4. Protection Criteria	3	3
4.1 General Criteria	3	3
4.2 Protection Criteria for Nuclear Safety-Related Systems and Components	3	3
4.3 Protection Criteria for Specific Systems and Components	4	4
5. Plant Hazards	5	5
5.1 Identification of Hazards	5	5
5.2 Factors Common to All Hazards	5	5
5.3 Discussion of Potential Hazards	5	5
6. Assessment of Need for Protection	8	8
6.1 Criteria for When Protection is Not Required	8	8
7. Protection Methods	9	9
7.1 Barriers and Enclosures	9	9
7.2 Restraints	9	9
7.3 Electrical Circuit Isolation	9	9
8. Implementation of Protection Methods	9	9
8.1 Conceptual Methods	9	9
8.2 Protection Against Specific Hazards	10	10
9. References	13	13
Appendices		
Appendix A	Example of Format for Tabulating Physical Protection Against Hazards to Nuclear Safety-Related Components	15
Appendix B	Plant Design Against Missiles	17
Figures		
Figure 1	Guide to Use of ANSI/ANS-58.3-1992	2
Figure 2	Protection of Functionally Redundant Systems	9
Figure 3	Protection of Reactor Coolant Pressure Boundary	10
Figure 4	Protection of Other Components	10
Tables		
Table 1	Examples of Hazards	6
Table A.1	Format for Tabulating Physical Protection Against Hazards	16
Table B.1	Missile Generation	20

Physical Protection for Nuclear Safety-Related Systems and Components

1. Introduction and Scope

1.1 Scope. This standard sets forth physical protection criteria for nuclear safety-related systems and components in stations using light water reactors (LWRs). This standard includes an identification of potential hazards to nuclear safety-related systems and components and acceptable means of ensuring the protection of this equipment from these hazards.

1.2 Purpose. The contents of this standard fall into two categories:

(1) **Criteria** – Criteria for which systems need protection are presented in Section 3, "Design Approach for Providing Protection" and Section 4, "Protection Criteria." Criteria for when protection is not needed are given in Section 6, "Assessment of Need for Protection."

(2) **Guidance** – Guidance and information useful to a designer who is faced with protecting systems and components from hazards are presented in Section 5, "Plant Hazards," Section 7, "Protection Methods," and Section 8, "Implementation of Protection Methods." The tutorial nature of some of the material in these sections reflects a desire to bring some important concepts to the attention of the designer.

1.3 Limits of Application of the Criteria. The designer shall implement the requirements of this standard through the use of other, more detailed, standards. For example, guidance to define one such hazard, namely pipe rupture, is addressed by American National Standard Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture, ANSI/ANS-58.2-1988 [1].¹

Guidance on protection of nuclear safety-related electrical equipment is found in American National Standard Criteria for Independence of Class 1E Equipment and Circuits, ANSI/IEEE 384-1981 [2] or its successor. It is neither desirable nor practical to list all possible codes and standards that may be applied to a particular plant design. It is the designer's responsibility to

locate the relevant guidance for each particular design.

An approach to design for protection is presented in Section 3, "Design Approach for Providing Protection," which recognizes the fact that a designer assesses protection of the entire system, including mechanical, structural, instrumentation and control, and electrical components. It is emphasized here that the designer assesses entire systems and perhaps groupings of systems in determining whether a hazard can cause unacceptable damage. The electrical designer assesses possible hazards due to mechanical systems interaction, such as pipe rupture and missiles as well as electrical hazards, as discussed in ANSI/IEEE 384 [2]; and the mechanical designer assesses the possibility of hazards in the electrical layout, such as collapse of nonseismically designed cable trays during an earthquake or transformer fires, as well as mechanical hazards. The structural designer determines whether certain accidents could overload or damage structures so that they could not fulfill their design requirements. Nuclear safety-related electrical or mechanical components in some cases may be located in nonseismic Category I structures. In such cases, the designer shall ensure that either the structure will not fail or that failure of the structure is considered as acceptable damage. Specific guidance on spatial separation criteria for electrical systems from other electrical systems is also available in ANSI/IEEE 384 [2].

Section 4, "Protection Criteria," presents protection criteria and discusses systems and components that need protection.

Section 5, "Plant Hazards," identifies the possible hazards to the plant against which protection shall be evaluated.

Section 6, "Assessment of Need for Protection," presents criteria that assist the designer in assessing the need for protection of nuclear safety-related systems and components.

Section 7, "Protection Methods," defines and discusses methods of achieving protection.

¹Numbers in brackets refer to corresponding numbers in Section 9, "References."

Section 8, "Implementation of Protection Methods," presents broad guidance on implementing protection methods.

2. Definitions

The definitions in this section are those unique to this standard. Definitions found in American National Standards Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, ANSI/ANS-51.1-1983(R1988), and Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants, ANSI/ANS-52.1-1983(R1988) [3,4], shall be utilized in this standard. The "Glossary of Terms in Nuclear Science and Technology" [5] and other referenced standards contain definitions of interest.

acceptable damage. Damage resulting from an event (or appropriate combinations of events) where the safety design requirements for the appropriate category of events are met. Events (and appropriate combinations of events) are categorized by plant conditions in ANSI/ANS-51.1 and ANSI/ANS-52.1 [3,4].

chemical attack. Detrimental influence due to chemical reaction.

harden. To strengthen against unfavorable environmental conditions.

residual heat. The total heat source remaining in a shut down reactor, including after-heat (i.e., heat from radioactive decay of fuel plus heat stored in structural components). (See "Glossary of Terms in Nuclear Science and Technology" [5].)

shall, should, and may. The word "shall" denotes a mandatory requirement. The word "should" denotes a recommendation. The word "may" denotes permission, neither a requirement nor a recommendation.

3. Design Approach for Providing Protection

Figure 1 is a flowchart of the progression that the designer shall follow to ensure that the plant design meets the protection criteria of this standard. The designer shall first identify those nuclear safety-related systems, structures, and components for which protection needs to be

assessed. The criteria for protection of these systems and components are given in Section 4, "Protection Criteria." Such identification shall address operational interfaces with other systems and redundant or diverse features. This identification shall clearly define system boundaries and the relationship of systems and components, either direct or indirect, with systems and components that are not nuclear safety-related.

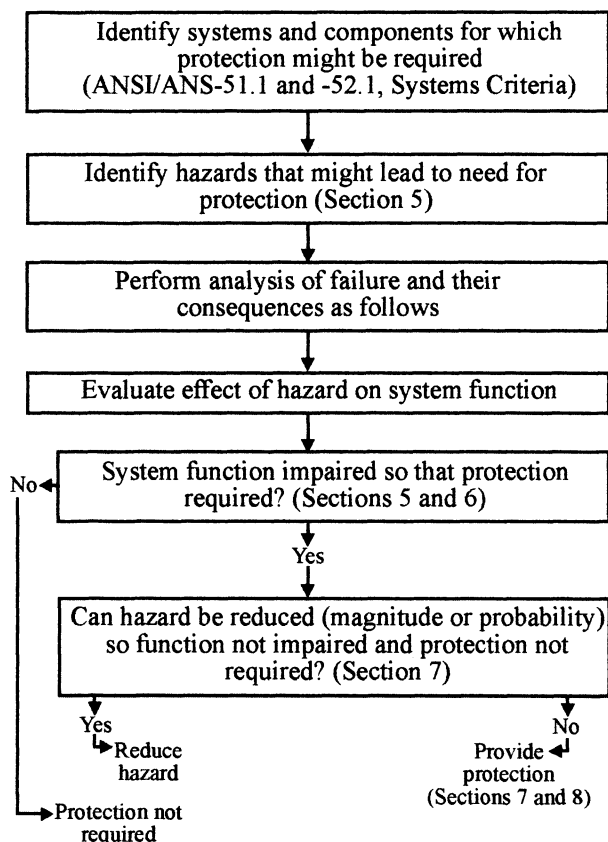


Figure 1
Guide to Use of ANSI/ANS-58.3-1992

In Section 4, "Protection Criteria," systems and components are discussed by function. However, specific systems, such as the residual heat removal system, are mentioned to provide illustrative examples of why protection may be required and how it may be accomplished. The protection criteria are treated in Section 4, "Protection Criteria," for the following three functional categories:

- (1) the reactor coolant pressure boundary,
- (2) the reactor containment, and
- (3) nuclear safety-related systems.

This standard does not provide the criteria to be used to define which systems or components are

nuclear safety-related or when redundancy or diversity is required. Those criteria are defined for LWRs in the systems criteria of ANSI/ANS-51.1 and -52.1 [3,4]. The designer shall refer to those criteria. Upon determination of those nuclear safety-related systems and components and their redundancy and diversity requirements for the plant, the designer shall then define the hazards against which protection shall be provided. Section 5, "Plant Hazards," identifies and discusses some of these hazards. In the identification of hazards, the designer shall assess events that are external to the system (such as fire, earthquake, or flooding), as well as events that could occur in a part of a system itself (such as rupture of one of the pipes in the system).

The designer shall develop a procedure to perform a detailed analysis of significant failures and their consequences utilizing this standard as a general guideline. Depending on the designer's preference, this may include the use of tables, matrices, or other aids.

The procedure shall require that a systematic analysis be performed of postulated events and that the result of that analysis be used to determine the effects on nuclear safety-related systems for the event being assessed.

The designer shall then evaluate the effect of each potential hazard on the identified nuclear safety-related systems and components and determine whether the damage is acceptable. This evaluation might require the use of other, more detailed standards, for example, ANSI/IEEE 384 [2]. Section 6, "Assessment of Need for Protection," lists options available to the designer upon completion of this evaluation. If the designer concludes that protection of a component against the hazard is required, Section 7, "Protection Methods," defines various protection methods that may be used. Further guidance is given in Section 8, "Implementation of Protection Methods," on the methods for providing protection for nuclear safety-related systems and components where the designer concludes such protection is required.

The method of evaluation shall be in such a form as to allow the analysis to be updated as necessary and to provide a record of the design basis for meeting protection requirements.² American

²An example of a form that might be used in the evaluation of physical protection against hazards for nuclear safety-related components is shown in Appendix A.

National Standard Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems, ANSI/IEEE 352-1987 [6], provides accepted principles and methods for reliability analysis. These design records shall be retained for the life of the plant and maintained as "living" documents. These evaluations shall be referenced when changes to the design are made throughout the life of the plant. It shall be ensured that the intent of the original design is not inadvertently negated by design changes, maintenance activity, or operations.

4. Protection Criteria

4.1 General Criteria. Protection shall be provided so that the hazards resulting from an event will not cause unacceptable damage to nuclear safety-related systems and components. (Reducing the probability of possible damage to a low enough level might be acceptable. See Section 6, "Assessment of Need for Protection." Nuclear safety-related systems and components are those that perform the nuclear safety functions necessary to ensure the following:

- (1) the integrity of the reactor coolant pressure boundary;
- (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or
- (3) the capability to prevent or mitigate the consequences of plant conditions that could result in potential off-site exposures that are comparable to the guideline exposures of Title 10, "Energy," Code of Federal Regulations (CFR), Part 100, "Reactor Site Criteria" [7].

Examples of nuclear safety-related systems include those having the following functions:

- (1) emergency core cooling;
- (2) emergency residual heat removal
- (3) containment isolation, heat removal, and combustible gas control;
- (4) containment atmosphere cleanup;
- (5) emergency electrical power;
- (6) reactor protection;
- (7) auxiliary support features (such as cooling) for nuclear safety-related components; and
- (8) post-accident habitability of the control room or other areas.

The nuclear safety-related systems for specific reactor types are discussed in more detail in their respective nuclear safety criteria standards (see Refs. [3] and [4]).

4.2 Protection Criteria for Nuclear Safety-Related Systems and Components. To be effective in performing nuclear safety-related

functions, the systems supplied to achieve the required nuclear safety functions (called "nuclear safety-related systems"), the auxiliary supporting systems necessary for their operation, and the associated protection and actuation systems required to initiate or apply the system function are required to all work in concert. Protection shall be provided for mechanical components, instrumentation and control components, and electrical components of any of these component systems whose successful performance is necessary to achieve the required nuclear safety function (see ANSI/IEEE-384 [2] for guidance on electrical equipment and protection methods). Monitoring equipment required for an operator to decide to take manual action essential to the introduction or continuation of a required nuclear safety function during the course of an accident or during post-accident control shall likewise be protected. In addition to protection of instrumentation against the hazard, instrumentation shall be designed to maintain the needed accuracy during the accident and post-accident decision-making process (see American National Standard Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors, ANSI/ANS-4.5-1980(1986) [8]).

4.3 Protection Criteria for Specific Systems and Components

4.3.1 Reactor Coolant Pressure Boundary (RCPB). Protection of the RCPB from hazards shall be provided so that

(1) No loss-of-coolant accident (LOCA) (see Ref. [3] or [4] for definition) for light water reactors shall occur as a result of a failure of any system, structure, or component not a part of the RCPB or as a result of any other event external to the RCPB.³

(2) No rupture of the RCPB that is not itself a design-basis (maximum-size) LOCA shall initiate a design-basis LOCA (e.g., the failure of one RCPB line shall not cause failure of another RCPB line such that the total combined break size results in a design-basis LOCA).³

(3) No rupture of the RCPB piping shall reduce the capability of the plant to less than the minimum required to protect the reactor core against the design-basis LOCA and to maintain a safe shutdown condition.³ See Ref. [3] or [4] for detailed guidance on coincident events.

4.3.2 Reactor Containment. The plant shall be designed so that the design-basis leak rate is

³See 6.1 (4) for acceptance of low probability events.

not exceeded unless it can be shown that the resultant off-site dose is within acceptable limits. Systems that, as a result of the event, are required to maintain the reactor containment function shall meet the system protection criteria of 4.3.3.

4.3.3 Nuclear Safety-Related Systems

4.3.3.1 Functionally Redundant Systems.

For those nuclear safety-related systems that are required to provide a nuclear safety function for the particular event, protection from the event or from consequences of the event shall be provided to maintain system function for the time period that system nuclear safety function is required.

As an example to illustrate the preceding case, assume that Systems A and B are redundant nuclear safety-related systems and that a single active failure of one system occurs.

For an event that requires the actuation of this system nuclear safety function, both Systems A and B shall be protected from the hazards associated with the event (see 8.1.1 and Figures 3 and 4). This is because the single active failure on one of the systems (either System A or System B) requires the remaining system to mitigate the consequences of the event.

4.3.3.2 Nonredundant Nuclear Safety-Related Components. A component that is not redundant but is nuclear safety-related, shall be protected from those events that could cause a loss of nuclear safety function of the component that is required to mitigate the event.⁴

4.3.3.3 Systems Containing Radioactivity. Protection shall be provided for inventories of stored radioactivity and for those portions of radioactive waste systems where the results of the event could result in off-site doses exceeding acceptable limits.⁵

4.3.3.4 System or Components Shared Between Nuclear Generating Units.⁶ Shared systems shall meet the requirements of Section 4, "Protection Criteria." In addition,

⁴This standard neither suggests exemption from, nor gives the criteria for, defining what systems or components require redundancy or diversity.

⁵The acceptable off-site radiation doses shall be within the limits as set forth in applicable federal regulations for the particular event. These limits may vary with both the probability of the event and the severity of its consequences. See Refs. [3] and [4].

⁶For guidance for shared emergency and shutdown electric systems for multi-unit nuclear power plants, see U.S. Nuclear Regulatory Commission Regulatory Guide 1.81, Revision 1, January 1975. (Regulatory Guides are available from the Superintendent of Documents, Government Printing Office, NRC, Washington, DC 20402).

(1) No event in one unit shall cause a reduction in the capacity of a shared nuclear safety-related system below that required to concurrently mitigate the consequences of the event in that unit, limit off-site radiation doses resulting from the event to acceptable limits, and bring and maintain all units to a safe shutdown condition.

(2) No event within a shared system shall prevent achieving and maintaining the safe shutdown of all units.

5. Plant Hazards

This section identifies some of the hazards that could be present inside or outside the plant for which protection of nuclear safety-related systems and components are to be assessed as required in Section 4, "Protection Criteria." Some of the events causing these hazards to the plant occur externally to the plant (e.g., earthquake, dam failure, and tornados that could create pressure, missiles, rain and debris blockage). Where external events could occur, protection of the nuclear safety-related systems from the resulting hazards shall be provided. Nuclear safety-related systems that are located external to the main plant buildings shall be protected also.

5.1 Identification of Hazards. Some of the hazards resulting from events that shall be assessed in designing for protection are as follows:

- (1) Missiles.
- (2) Pressure.
- (3) Temperature.
- (4) Pipe whip.
- (5) Fluid jet.
- (6) Fire.
- (7) Radiation.
- (8) Steam or humidity.
- (9) Chemical attack.
- (10) Flooding.
- (11) Debris blockage.

5.2 Factors Common to All Hazards. In determining the protection requirements for components, the following factors shall be addressed:

- (1) The source, magnitude, and duration of the hazard.
- (2) The geometry or physical configuration of the hazard, the component to be protected, and the structures in the area.
- (3) The potential for propagation of the hazard into other areas or the potential for the hazard to initiate other events.

(4) The protection afforded against the hazard by installed protection systems, operator action alone, or some combination thereof.

(5) Possible secondary effects of the hazard (e.g., spalling due to missile impact, flooding due to actuation of fire control equipment, flooding due to a failure or malfunction of a fluid system, automatic actuation of other systems, or nonaccessibility to an area for which access is required).

5.3 Discussion of Potential Hazards

5.3.1 Missiles. A discussion of missiles, as well as energy sources capable of causing missiles, has been formulated. These energy sources include, but are not limited to, the following:

- (1) Fluid energy.
- (2) Mechanical kinetic energy.
- (3) Mechanical strain energy.
- (4) Chemical energy.
- (5) Gravitational potential energy.
- (6) Energy of natural phenomena.
- (7) Energy of transport devices.
- (8) Electrical energy.

For more details, see ANSI/IEEE-384 [2]; American National Standard Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites, ANSI/ANS-2.3-1983 [9]; and American National Standard Guidelines for Combining Natural and Man-Made Hazards at Power Reactor Sites, ANSI/ANS-2.12-1978 [10].⁷

5.3.2 Pressure and Temperature. Pipe rupture, explosion, or safety valve discharge are examples of events that could cause increased pressure and temperature in an enclosure and increased differential pressure across walls and other physical barriers.

Increased pressure of high enough value to cause failure of a wall or slab could lead to loss of nuclear safety function of a component. This indirect effect of pressure is usually a more important design parameter than the effect of pressure on the component itself. Increased external pressure could cause direct failure of the component if the component is not designed for the pressure acting on its exterior (e.g., large tanks, ventilation equipment, or diaphragm valve operators). Differential pressure could also be created by events that cause vacuum in systems, components, or buildings.

⁷See Appendix B for additional guidance.