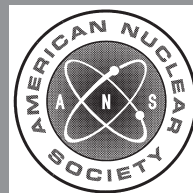


American Nuclear Society

**safety and pressure integrity
classification criteria for
light water reactors**

an American National Standard



published by the
American Nuclear Society
555 North Kensington Avenue
La Grange Park, Illinois 60526 USA

**American National Standard
Safety and Pressure Integrity
Classification Criteria for
Light Water Reactors**

Secretariat
American Nuclear Society

Prepared by the
**American Nuclear Society
Standards Committee
Working Group ANS-58.14**

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

Approved April 22, 2011
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 It may be reaffirmed, revised, or withdrawn at any time in accordance with established procedures.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 © 2011 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-58.14-2011 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

Inquiry Requests

The American Nuclear Society (ANS) Standards Committee will provide responses to inquiries about requirements, recommendations, and/or permissive statements (i.e., “shall,” “should,” and “may,” respectively) in American National Standards that are developed and approved by ANS. Responses to inquiries will be provided according to the Policy Manual for the ANS Standards Committee. Nonrelevant inquiries or those concerning unrelated subjects will be returned with appropriate explanation. ANS does not develop case interpretations of requirements in a standard that are applicable to only a specific design, operation, facility, or other unique situation and therefore is not intended for generic application.

Responses to inquiries on standards are published in ANS’s magazine, *Nuclear News*, and are available publicly on the ANS Web site or by contacting the ANS standards administrator.

Inquiry Format

Inquiry requests must include the following:

- (1) the name, company name if applicable, mailing address, and telephone number of the inquirer;
- (2) reference to the applicable standard edition, section, paragraph, figure, and/or table;
- (3) the purposes of the inquiry;
- (4) the inquiry stated in a clear, concise manner;
- (5) a proposed reply, if the inquirer is in a position to offer one.

Inquiries should be addressed to

American Nuclear Society
ATTN: Standards Administrator
555 N. Kensington Avenue
La Grange Park, IL 60526
or standards@ans.org

Foreword

(This Foreword is not a part of American National Standard “Safety and Pressure Integrity Classification Criteria for Light Water Reactors,” ANSI/ANS-58.14-2011.)

This standard revises and supersedes the safety and pressure integrity classification criteria provided in ANSI/ANS-51.1-1983 (R1988) (withdrawn), “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants,” and ANSI/ANS-52.1-1983 (R1988) (withdrawn), “Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants.”¹⁾ The criteria in this standard are primarily objective; are applicable to all nuclear power plant functions, structures, systems, components, and parts (including consumables); and are applicable to any light water reactor (LWR) nuclear power plant design. The criteria in ANSI/ANS-51.1-1983 (R1988) (withdrawn) and ANSI/ANS-52.1-1983 (R1988) (withdrawn) are primarily subjective, apply primarily to systems, and apply only to the new designs of pressurized water reactors and boiling water reactors (BWRs) available in the United States in 1983.

This standard uses separate sets of terms for safety classification criteria²⁾ and pressure integrity classification criteria.³⁾ ANSI/ANS-51.1-1983 (R1988) (withdrawn) and ANSI/ANS-52.1-1983 (R1988) (withdrawn) address both safety and pressure integrity classification criteria using a single set of terms (Safety Classes 1, 2, and 3, and Non-Nuclear Safety). The applicability of these two sets of criteria is not identical. The single set of terms used in ANSI/ANS-51.1-1983 (R1998) (withdrawn) and ANSI/ANS-52.1-1983 (R1988) (withdrawn) creates inconsistencies and a potential for misinterpretations. These limitations are avoided in this standard.

The safety classification criteria in this standard are based on NEDC-31509, “Safety Classification Methodology and Criteria for Structures, Systems, Components and Parts in BWR Nuclear Power Plants,” developed by the Parts Safety Classification Committee of the BWR Owners Group and GE Nuclear Energy, and on EPRI NP-6895, “Guidelines for the Safety Classification of Systems, Components and Parts Used in Nuclear Power Plant Applications (NCIG-17),” developed by the Nuclear Construction Issues Group, a utility group sponsored by the Electric Power Research Institute.

The application of many requirements to nuclear power plant structures, systems, components, and parts is based upon their safety classification. The safety classification of an item is typically used to determine which design, procurement, manufacturing, construction, and operating requirements or controls apply.

The term “safety-related” is used to identify items that, because of their functional safety importance, must meet stringent design requirements such as Seismic Category I criteria; IEEE Class 1E criteria for electrical items; *ASME Boiler and Pressure Vessel Code*, Sec. III, criteria for pressure integrity items; and environmental qualification requirements of *Code of Federal Regulations*, Title 10, “Energy,” Part 50, “Domestic Licensing of Production and Utilization Facilities,” Sec. 49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants.”

The safety classification of an item might be also used to help establish the procurement requirements for the item. Typically, there are three types of procurement classifications: safety-related, commercial grade, and non-safety-related.

¹⁾ The remaining portions of ANSI/ANS-51.1-1983 (R1988) (withdrawn) and ANSI/ANS-52.1-1983 (R1988) (withdrawn) are not superseded by this standard.

²⁾ Safety-related (Q), non-safety-related with augmented quality assurance [or augmented (A)], and non-safety-related (N).

³⁾ Classes 1, 2, 3, 4, and 5.

A safety-related procurement refers to an item that is purchased subject to the provisions of *Code of Federal Regulations*, Title 10, “Energy,” Part 21, “Reporting of Defects and Noncompliance” (10 CFR 21), and is intended for use in applications that are functionally safety-related. Commercial-grade procurement refers to an item that is purchased without the provisions of 10 CFR 21 but is intended to be dedicated after receipt for use in applications that are functionally safety-related. Once a commercial-grade item is dedicated, it becomes a safety-related item. Non-safety-related procurement refers to an item that is purchased without the provisions of 10 CFR 21 and is intended for use in applications that are functionally non-safety-related.

During construction, safety-related items are subject to specific material selection, design, fabrication, examination, testing, inspection, certification, installation, and quality assurance requirements.

Operationally, safety-related items typically are subject to specified requirements for in-service inspection, in-service testing, maintenance, surveillance, and quality assurance.

The classification “non-safety-related with augmented requirements” is applied to certain non-safety-related items during procurement, construction, and operations when the item is not safety-related but is relied upon during a special event or where licensing requirements exist.

The focus of safety classification in this standard is on the accomplishment of safety-related functions that may be considered to provide design-basis-event prevention, mitigation, or both, without emphasizing one over the other. Multiple redundant levels of defense provide a balance of defense-in-depth strategies such that no single element (e.g., accident prevention) or barrier (e.g., containment) is emphasized to the exclusion of others. An adequate balance of prevention and mitigation as well as consideration of defense in depth is provided in the classification process through the consideration of the three basic safety-related functions identified in *Code of Federal Regulations*, Title 10, “Energy,” Part 50, “Domestic Licensing of Production and Utilization Facilities,” Sec. 2, “Definitions.”

The pressure integrity classification criteria provided in Sec. 5 are similar to those of ANSI/ANS-51.1-1983 (R1988) (withdrawn), ANSI/ANS-52.1-1983 (R1988) (withdrawn), and Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” but have been revised to be applicable to any LWR design (particularly an advanced passive design).

The basic design requirements for items assigned to each safety classification are summarized in Sec. 6.

This standard has been written for prospective use, but the criteria are based on current practices and requirements applicable to licensed LWR designs.

This standard might reference documents and other standards that have been superseded or withdrawn at the time the standard is applied. A statement has been included in the references section (Sec. 7) that provides guidance on the use of references.

This standard does not incorporate the concepts of generating risk-informed insights, performance-based requirements, or a graded approach to quality assurance. The user is advised that one or more of these techniques could enhance the application of this standard.

Working Group ANS-58.14 of the Standards Committee of the American Nuclear Society had the following membership at the time it developed this standard:

M. A. Linn (Chair), *Oak Ridge National Laboratory*

D. P. Blanchard, *Applied Reliability Engineering, Inc.*

S. A. Highley, *AREVA NP, Inc.*

R. A. Hill, *ERIN Engineering and Research, Inc.*

G. B. Locklear, *Engineering & Management Specialists, Inc.*

P. A. Sicard, *Entergy*

R. B. Williston, *Individual*

Subcommittee ANS-22, Systems Design Criteria, had the following membership at the time of its approval of this standard:

D. G. Newton, (Chair), *AREVA NP, Inc.*

R. M. Ruby (Vice Chair), *Constellation Energy*

N. W. Brown, *Individual*

R. A. Hill, *ERIN Engineering and Research, Inc.*

L. E. Kreider, *Engineering Planning & Management, Inc.*

M. A. Linn, *Oak Ridge National Laboratory*

D. W. Murphy, *Bechtel Corporation*

The Nuclear Facilities Standards Committee (NFSC) had the following membership at the time of its approval of this standard:

C. A. Mazzola (Chair), *Shaw Environmental & Infrastructure, Inc.*

R. M. Ruby (Vice Chair), *Constellation Energy Company*

J. A. August, *CORE, Inc.*

W. H. Bell, *South Carolina Electric & Gas Company*

J. R. Brault, *Shaw MOX Project*

C. K. Brown, *Southern Nuclear Operating Company*

K. R. Bryson, *Shaw Environmental, Inc.*

C. E. Carpenter, *U.S. Nuclear Regulatory Commission*

(Alt. A. Hull, *U.S. Nuclear Regulatory Commission*)

D. R. Eggett, *Automated Engineering Services Corporation*

R. W. Englehart, *Individual*

P. K. Guha, *U.S. Department of Energy*

(Alt. J. O'Brien, *U.S. Department of Energy*)

P. S. Hastings, *Duke Energy Company (NuStart Liaison)*

R. A. Hill, *ERIN Engineering and Research, Inc.*

N. P. Kadambi, *Individual*

E. M. Lloyd, *Exitech Corporation*

S. A. Lott, *Los Alamos National Laboratory*

R. H. McFetridge, *Westinghouse Electric Company, LLC*

(Alt. L. R. Grobmyer, *Westinghouse Electric Company, LLC*)

T. K. Meneely, *Westinghouse Electric Company, LLC*

(Alt. R. C. Surman, *Westinghouse Electric Company, LLC*)

C. H. Moseley, *ASME NQA Liaison*

D. G. Newton, *AREVA NP*

W. N. Prillaman, *AREVA NP*

W. B. Reuland, *Individual*

J. C. Saldarini, *Bechtel SAIC Company, LLC*

(Alt. A. T. Vieira, *Bechtel Power Corporation*)

D. J. Spellman, *Oak Ridge National Laboratory (NFSC Liaison to IEEE NPEC)*

S. L. Stamm, *Shaw Nuclear Services*

J. D. Stevenson, *Individual*

J. A. Wehrenberg, *Southern Company Services*

M. J. Wright, *Entergy Operations, Inc.*

NFSC Liaison:

G. Hutcherson, *Institute of Nuclear Power Operations*

J. H. Riley, *Nuclear Energy Institute*

(Alt. J. C. Butler, *Nuclear Energy Institute*)

NFSC Observers:

R. H. Bryan, *Tennessee Valley Authority*

E. P. Loewen, *General Electric*

J. E. Love, *Bechtel Power Corporation*

C. D. Thomas, *Individual*

R. E. Scott, *Individual*

Contents	Section	Page
1	Introduction	1
1.1	Scope	1
1.2	Purpose	1
1.3	Applicability	1
2	Acronyms and definitions	2
2.1	List of acronyms	2
2.2	Definitions	2
3	General requirements	6
3.1	Regulatory basis	6
3.2	Relationship of functional, procurement, and application safety classifications	7
3.3	Methodology	7
4	Safety classification criteria	9
4.1	General criteria	9
4.2	Determination of design basis events	11
4.3	Determination of safety-related functions	11
4.4	Determination of safety-related systems and structures	12
4.5	Determination of safety-related components and parts	13
4.6	Items not classified safety-related	21
5	Pressure integrity classification criteria	23
5.1	Functional criteria	23
5.2	Interface criteria	24
6	Basic design requirements	25
6.1	Pressure-retaining items	26
6.2	Electrical items	26
6.3	Seismic classification	26
6.4	Environmental qualification	26
6.5	Quality assurance	26
7	References	26
Appendices		
Appendix A	Relationship of the Terms “Safety-Related” and “Safety”	30
Appendix B	Classification Categories	32
Appendix C	Safety Classification Examples	36
Appendix D	Typical Examples of Interfaces Between Pressure Integrity Classes	60
Tables		
Table 1	Basic design requirements	25
Table A.1	Approximate relationship of various safety classification terms	30
Figures		
Figure 1	Methodology for safety classification	7
Figure 2	Safety-related/non-safety-related fluid system boundary criteria	15

Figure 3 Boundary criteria for fluid system lines penetrating primary containment 16

Figure 4 Boundary criteria for instrument lines connected to the RCPB and penetrating primary containment 19

Figure 5 Boundary criteria for instrument lines not penetrating primary containment (Note 1) 20

Figure C.1 Auxiliary feedwater system (examples 2 and 3) 39

Figure C.2 Battery room heat removal system (examples 4 and 5) 43

Figure C.3 Auxiliary power system synchronizing circuit (example 8) .. 47

Figure C.4 Hydraulic snubber seals (example 21) 58

Figure D.1 Typical fluid-system pressure integrity class interfaces ... 61

Safety and Pressure Integrity Classification Criteria for Light Water Reactors

1 Introduction

This standard provides criteria for classification of structures, systems, and components (SSCs) for light water reactors (LWRs). The standard addresses criteria for classification from both a safety and pressure-retaining perspective for licensing design basis events (DBEs). The criteria are largely deterministic in nature, without explicit consideration of probabilities or the magnitude of consequences other than the application of dose limits as criteria for DBEs. However, the deterministic approach contains implied elements of probability, from the selection of accidents to be analyzed to system-level requirements [1].¹⁾

In this standard, the categories with respect to safety classification are denoted by the terms “safety-related,” “non-safety-related with augmented requirements,” and “non-safety-related” and by Classes 1, 2, 3, 4, and 5 for pressure integrity classification. The bases for this terminology are discussed in other parts of this document.²⁾

The basic design requirements for each safety classification are discussed in Sec. 6.

1.1 Scope

This standard specifies criteria for the safety classification of items (SSCs and parts, including consumables) in a LWR nuclear power plant as safety-related (Q), non-safety-related with augmented requirements (A), or non-safety-related (N). In addition, pressure integrity classification criteria are provided for the assignment of Classes 1, 2, 3, 4, or 5 to pressure-retaining³⁾ items.

1.2 Purpose

This standard has two purposes. The first is to provide criteria for the safety classification of items in LWR nuclear power plants. The second is to provide criteria for the assignment of pressure integrity classes to pressure-retaining items.

The application of the criteria specified in this document is intended to ensure that the safety and pressure integrity classifications of an item are based on its functions.

The classification criteria specified in this standard

- (1) establish an objective, technically sound rationale for the determination of safety and pressure integrity classifications;
- (2) include provisions for ensuring that all items required by the licensing design basis are appropriately classified;
- (3) allow maximum compatibility with plant-specific designs, policies, and procedures;
- (4) achieve consistency (i.e., minimize classification differences among similar items in the plant) among utilities, vendors, designs, and regulators, and with current industry and regulatory practices.

1.3 Applicability

This standard is applicable to LWR designs. It is based upon current regulations, criteria, and experience. Some of the criteria and guidance provided herein might differ from those used as the bases for previously licensed designs.

¹⁾ Numbers in brackets refer to corresponding numbers in Sec. 7, “References.”

²⁾ See the Foreword and Appendices A through D for additional discussion.

³⁾ The scope of pressure-retaining items with respect to this standard is discussed in Sec. 5.

2 Acronyms and definitions

2.1 List of acronyms

A: non-safety-related with augmented requirements

ALWR: advanced light water reactor

AST: alternate source term

ANS: American Nuclear Society

ANSI: American National Standards Institute

ASME: American Society of Mechanical Engineers

ATWS: anticipated transient without scram

B&PVC: *ASME Boiler and Pressure Vessel Code*

BWR: boiling water reactor

CFR: *Code of Federal Regulations*

CRD: control rod drive

DBE: design basis event

ECSS: emergency core cooling system

EOP: emergency operating procedure

EPRI: Electric Power Research Institute

ESF: engineered safety feature

FSAR: Final Safety Analysis Report

IEEE: Institute of Electrical and Electronics Engineers

ISA: International Society of Automation

LBD: licensing basis documentation

LOCA: loss-of-coolant accident

LWR: light water reactor

N: non-safety-related

NRC: U.S. Nuclear Regulatory Commission

P&ID: piping and instrumentation drawing

PWR: pressurized water reactor

Q: safety-related

RCPB: reactor coolant pressure boundary

RCS: reactor coolant system

RTNSS: regulatory treatment of non-safety systems

RTS: reactor trip system

SBO: station blackout

SSAR: standard safety analysis report

SSCs: structures, systems, and components

SSE: safe shutdown earthquake

TIP: traversing in-core probe

2.2 Definitions

This section defines terms whose meanings are specific to this standard.

active function: A function where mechanical motion, actuation, or a change of state occurs (e.g., the closing of a valve or relay or the change in state of a logic circuit).⁴⁾

component: An item that performs a specific function within a system (usually has a component-level plant-unique identification code) and can be either an assembly of interconnected parts or a single part.

consumables: Materials or supplies consumed or expended during installation, operation, testing, maintenance, or repair of plant structures, systems, components, or parts, such as

- items consumed or expended during the operation of components or routinely replaced during maintenance (e.g., diesel fuel, O-rings, gaskets, hydraulic fluid, lubricating oil, grease, packing, and paint);
- items consumed or expended in maintaining the chemical control of system process fluids (e.g., resins, additive chemicals and gases such as boron, pH buffer, bromophenol blue, and nitric acid); or
- items consumed or expended during maintenance, installation, and modification activities and that are generally used throughout the plant and might not be

⁴⁾ In passive advanced light water reactor (ALWR) designs, passive engineered safety features (ESFs) are those that rely on passive means to provide their functions. Passive means are natural forces (e.g., gravity and natural circulation), stored energy (e.g., batteries, rotating inertia, and compressed fluid), energy inherent to the system, and noncycling valves. Passive means do not rely on large, continuously rotating machinery; multiple acting valves; or alternating-current-powered divisions of electrical power.

included in the above (e.g., solvents, layout fluid, welding rods, leak-testing fluid, tape, and penetrant testing materials).

design basis event (DBE): An event that is a condition of normal operation, including anticipated operational occurrences, design basis accidents (or transient), external events, or natural phenomena for which the plant must be designed to ensure the three basic safety-related functions are achieved. See the definition for “safety-related” for a description of the three basic safety-related functions.

design function: A principal function for which a SSC was included in the plant design. Design functions, as they relate to safety-related items, support one or more of the three basic safety-related functions. For example, the emergency core cooling system (ECCS) is included in the plant design to perform the primary design function of providing coolant to the reactor vessel during or following a loss-of-coolant accident (LOCA), in order to maintain the unit in a safe shutdown condition.

engineered safety feature (ESF): A SSC that is relied upon during or following a DBE to ensure the capability to prevent or mitigate the consequences of that DBE that could result in potential off-site exposures comparable to the guideline exposures in applicable regulations,⁵⁾ excluding reactor coolant pressure boundary (RCPB) and reactor trip system (RTS) items.

failure: The inability of an item to accomplish its design function.

item: Any plant structure, system, component, or part (including consumables).

licensing basis documentation (LBD): The set of documents that specify the licensing requirements and commitments that form the basis used by the U.S. Nuclear Regulatory Commission (NRC) to license a nuclear power plant or a standard plant design. For example, the LBD may include

- (1) the latest version of either the Final Safety Analysis Report (FSAR) or the Standard Safety Analysis Report (SSAR);
- (2) NRC safety evaluation reports;

(3) operating license, combined license, final design approval, or design certification, including the technical specifications;

(4) correspondence between NRC and the licensee that contains licensing requirements or commitments for the design or operation of the nuclear power plant or standard plant design.

licensing commitment: A commitment specified in the plant LBD (e.g., a commitment to apply specific design criteria to an item or to implement the licensing guidance provided by NRC in a Generic Letter or Regulatory Guide).

licensing guidance: An NRC recommendation of an acceptable way to comply with a licensing requirement (e.g., regulatory guides, standard review plan, regulatory issue summaries, etc.).

licensing requirement: A regulation or a requirement stemming directly or indirectly from a regulation established by NRC (e.g., the *Code of Federal Regulations*, a license, or order).

loss-of-coolant accident (LOCA): The loss of reactor coolant, at a rate in excess of the reactor coolant normal makeup rate, from breaks or openings in the RCPB, up to and including a break equivalent in size to the largest pipe rupture justified in 10 CFR 50 Appendix K [2]—or, in the absence of justification, a double-ended rupture of the largest pipe—in the RCPB. (See 10 CFR 50, Appendix A [3], or Regulatory Guide 1.157, Position 3.1) [4].

nondetectable failures: Failures that cannot be identified through periodic testing or revealed by alarm or anomalous indication. Assuming adequate design and quality assurance programs, training, control room design, and procedures, nondetectable failures do not include common-mode failures resulting from failures due to external environmental effects or design deficiencies or errors due to manufacturing, maintenance, or operations [5].

non-safety-related with augmented requirements (A): Classification applied to an item that is not safety-related but that is relied upon

⁵⁾ Refer to the definition of “safety-related” in Sec. 2.2 for further information regarding “guideline exposures in applicable regulations.”

during a special event or to which a licensing requirement or commitment applies.⁶⁾

non-safety-related (N): Classification applied to an item that is not safety-related.

part: A basic element of a SSC that ordinarily cannot be, or would not be, disassembled further for procurement or maintenance purposes (and might have a part-level plant-unique identification code).

passive function: A function that is not an active function (e.g., the pressure-retaining function of a valve that is not required to change its position or an instrumentation and control component that is not required to change output state).

plant-level safety-related function: Plant-level safety-related functions are functions performed by a system that directly supports one or more of the three basic safety-related functions during a DBE. For example, the ECCS provides the plant-level safety-related function of reactor inventory control during a LOCA, which is a safe shutdown function.⁷⁾

primary containment: The principal enclosure that acts as a barrier, after the RCPB, to control the release of radioactive material from the fuel in the reactor core. This boundary includes the following:

- (1) the primary containment structure and its access closures, penetration closures, and appurtenances;
- (2) those valves, pipes, closed systems, and other pressure-retaining components used to effect isolation of the primary containment atmosphere from the external environment;
- (3) those systems or portions of systems that, by their system functions, extend the primary containment boundary (e.g., the connecting steam and feedwater piping and isolation valves).

reactor coolant pressure boundary (RCPB): All pressure-containing components of LWR nuclear power plants, such as pressure vessels, piping, pumps, and valves, that are either

(1) part of the reactor coolant system (RCS); or

(2) connected to the RCS up to and including any or all of the following:

- (a) the outermost primary containment isolation valve in system piping that penetrates the primary containment,
- (b) the second of two valves normally closed during normal reactor operation in system piping that does not penetrate primary containment, or
- (c) the RCS safety and relief valves.

For a direct-cycle boiling water reactor (BWR), the RCS extends to and includes the outermost primary containment isolation valve in the main steam and feedwater piping. (See 10 CFR 50.2 [6].)

reactor coolant system (RCS): The system consisting of those items relied upon to (a) form the pressure-retaining boundary needed to contain the reactor coolant, (b) transfer heat during normal operation from the reactor core to the power conversion system or to the shutdown heat removal systems, and (c) provide system pressure relief.

reactor trip system (RTS): The principal system credited in the LBD for shutdown of the reactor.

relied upon: Dependence on the successful functioning of the SSCs in the following manner:

- (1) for DBEs: the SSCs that are necessary and sufficient (i.e., the minimal subset of SSCs) to assure each of the three basic safety-related functions in the presence of a postulated single failure plus any other coincident occurrence assumed in the safety analysis;
- (2) for special events: the SSCs that are necessary and sufficient (i.e., the minimal subset of SSCs) to assure required functions in the presence of postulated failures (e.g., common cause) related to the specific special event.

⁶⁾ See Sec. 4.6.1 for identification of items that are typically classified A.

⁷⁾ Plant-level, safety-related functions are defined by the licensee and can be similar to those developed for other regulatory purposes [such as development of emergency operating procedures (EOPs)].

safe shutdown: A unit shutdown with (a) the reactivity of the reactor kept to a margin below criticality, consistent with technical specifications; (b) the core decay heat removed at a controlled rate, sufficient to prevent core or RCS thermal design limits from being exceeded⁸⁾; (c) radioactive material releases controlled to keep doses within prescribed limits; and (d) operation within design limits of SSCs necessary to maintain these conditions.

safety-related: Applies to a function, SSC, or part that is relied upon during or following a DBE to ensure these functions⁹⁾:

- (1) the integrity of the RCPB;
- (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 accidents that could result in potential off-site exposures comparable to the guideline exposures in 10 CFR 50.34(a)(1) [7], 10 CFR 50.67(b)(2) [8], or 10 CFR 100.11 [9], as applicable.¹⁰⁾

This is the regulatory definition of “safety-related” [6, 11, 12].

To avoid confusion, only the term “safety-related” is used in this standard. The term “safety” is not to be used as a synonym for “safety-related” but is to be used in a broader sense only when necessary.¹¹⁾

safety-related (Q): Classification applied to

- (1) an item relied upon to remain functional during or following a DBE to ensure a safety-related function;
- (2) documented information that specifies or establishes parameters required to ensure a safety-related function of an item; or

(3) services (to design, purchase, fabricate, handle, ship, store, clean, erect, install, inspect, test, operate, maintain, repair, refuel, or modify) that ensure a safety-related function of an item.

secondary containment: The structure or structures that act as a barrier, after the primary containment, so that radioactive material leakage from the primary containment or ESFs during or following a DBE can be collected and processed before release to the external environment. The structure or structures might include the reactor building, access closures, and penetrations of systems that directly communicate with the secondary containment atmosphere.

Seismic Category I: The classification of an item that is designed to perform at least one safety-related function during or following a safe shutdown earthquake (SSE).¹²⁾

Seismic Category II: The classification of an item that is not Seismic Category I but whose failure mode could prevent a safety-related function during or following a SSE.¹²⁾

shall, should, and may: The word “shall” denotes a requirement; the word “should” denotes a recommendation; and the word “may” denotes permission, neither a requirement nor a recommendation.

shut down the reactor: Insert negative reactivity into the reactor core to make it subcritical.

single failure: A random failure and its consequential effects, in addition to an initiating occurrence, that result in the loss of capability of a component to perform its intended safety-related function (or functions). Fluid and electrical systems are considered to be designed against an assumed single failure if neither (a) a single failure of any active component (assuming passive components function properly)

⁸⁾ Sufficient core decay heat removal does not imply the need for cold shutdown.

⁹⁾ Parts (1), (2), and (3) are the three basic safety-related functions.

¹⁰⁾ Dose guidelines of 10 CFR 100.11 are superseded by 10 CFR 50.67 for facilities that have implemented an alternate source term (AST). See Regulatory Guide 1.183 [10].

¹¹⁾ The Institute of Electrical and Electronics Engineers (IEEE) definition of “safety” is identical to “safety-related” (see ANSI/IEEE-603-1998 [13]). The term “Class 1E” refers to the electrical portions of safety-related systems and equipment as defined by IEEE. See Appendix A for additional discussion.

¹²⁾ See Electric Power Research Institute (EPRI), “Advanced Light Water Reactor Utility Requirements Document” [14], and Regulatory Guide 1.29 [15].

nor (b) a single failure of any passive component (assuming active components function properly) results in a loss of the capability of the system to perform its safety-related function.¹³⁾

special event: An event that is part of the plant licensing basis but is not a DBE. Special events are identified in the plant LBD and typically include anticipated transients without scram (ATWS), fire, and station blackout (SBO).

structures: Civil constructions (e.g., buildings), portions of civil constructions (e.g., doors, hatches, or walls), supports for items, or structural components such as fuel racks and building sump debris screens. (These might have system-level or component-level plant-unique identification codes).

system: A group of physically interconnected components that together perform a specified design function (or functions). (Systems are normally designated by a system-level plant-unique identification code.)

3 General requirements

This section provides an overview of the safety classification process. Included is the regulatory basis for the safety and pressure integrity classification criteria and the relationships among various classification terms.

The safety classification process starts with the definition of DBEs documented for the plant and extends sequentially down to the safety-related plant-level functions, systems, and components and eventually to parts (including consumables). Thus, the safety classification of a part depends on the classification of items higher in the process. This process identifies the necessary and sufficient set of safety-related items that satisfies the regulatory definition.

Pressure integrity classification is a prerequisite for the specification of design, quality, in-service inspection, and in-service testing requirements of applicable fluid systems.

3.1 Regulatory basis

Pertinent requirements of 10 CFR 50, Appendix B [18] apply to activities¹⁴⁾ affecting the safety-related functions of SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Identification of these SSCs is required by 10 CFR 50, Appendix B, Criterion II.¹⁵⁾ NRC regulations 10 CFR 50.49(b)(1) and 10 CFR 100, Appendix A, VI(a)(1), together define SSCs that will be environmentally and seismically qualified. A basic component to which a quality assurance program is to be applied is similarly defined in 10 CFR 21.3(1)(i) [20]. These regulations establish the following three basic safety-related functions for which identification of safety-related SSCs relied upon during DBE is required:

- (1) the integrity of the RCPB;
- (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 accidents that could result in potential off-site exposures comparable to the guideline exposures in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

The safety classification steps and criteria provided in Secs. 3.3 and 4, which define SSCs that shall be classified as safety-related, identify in greater detail the plant functions, structures, systems, and components that achieve the three basic safety-related functions. The classifications in Secs. 3.3 and 4 are based upon, and directed at implementing, these regulations.

¹³⁾ ANSI/ANS-51.1-1983 (R1988) (withdrawn) [16] and ANSI/ANS-52.1-1983 (R1988) (withdrawn) [17].

¹⁴⁾ Although the focus of this standard is the classification of SSCs, classification is closely tied to the treatment prescribed depending on the classification. Some of these “activities” (treatment) can be modified from a traditional, deterministic approach by a risk-informed approach as described in 10 CFR 50.69 [19].

¹⁵⁾ Note that application of 10 CFR 50, Appendix B, does not imply a safety-related classification.

Assignment of different classes of pressure integrity criteria to pressure-retaining items is required to apply quality group standards per 10 CFR 50.55a [21].¹⁶⁾

3.2 Relationship of functional, procurement, and application safety classifications

The functional classification (safety-related, non-safety-related with augmented requirements, or non-safety-related) is assigned to an item based on the functions performed by the item. This classification (safety-related, non-safety-related with augmented requirements, or non-safety-related) applies to an item during procurement [or following procurement if commercial-grade dedication is performed¹⁷⁾], during and following installation of the item into the plant, and is based on the licensing requirements or commitments that are applicable to the item once installed.

The relationship of safety classifications to other classification categories (such as pressure integrity classification or seismic classification) is discussed in Sec. 6.

3.3 Methodology

The steps provided in Sec. 3.3 summarize the methodology that shall be used to classify functions, structures, systems, components, and parts (including consumables) as safety-related (Q), non-safety-related with augmented requirements (A), or non-safety-related (N). These steps are outlined in Fig. 1. The criteria specified in Sec. 4 shall be used to perform these steps. Items may be classified on an as-needed basis, e.g., during design, maintenance, or replacement. Items that have not been classified may be designated unclassified.

Note that only those SSCs necessary and sufficient to accomplish any of the three basic safety-related functions need to be classified as safety-related. This may mean that not all SSCs capable of performing a given safety-related function need to be classified as safety-related. In addition, classification of SSCs in their support of the three basic safety-related functions considers the presence of the most limiting sin-

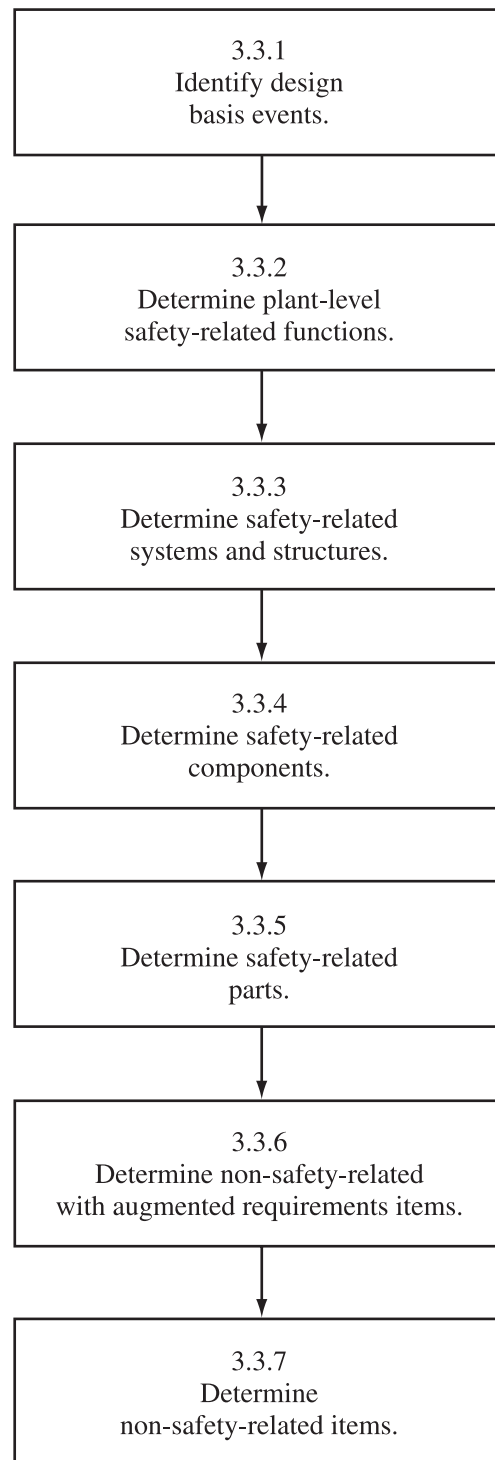


Figure 1 – Methodology for safety classification

¹⁶⁾ See also Regulatory Guide 1.26 [22].

¹⁷⁾ Commercial-grade procurement refers to an item that is purchased without the provisions of 10 CFR 21 but is intended to be dedicated after receipt for use in applications that are functionally safety-related. Once a commercial-grade item is dedicated, it becomes a qualified safety-related item.

gle failure plus any coincident occurrences assumed in the safety analysis for a given DBE.

3.3.1 Identify design basis events.

All DBEs shall be identified (see Sec. 4.2).

3.3.2 Determine plant-level safety-related functions.

The plant-level safety-related functions [e.g., reactivity control, RCS integrity, RCS heat removal, RCS inventory control, containment control, etc.¹⁸⁾] relied upon in the safety analyses of DBEs to perform any of the three basic safety-related functions shall be identified.

Plant-level safety-related functions that prevent or mitigate DBEs whose consequences could result in potential off-site exposures (as described in the DBE analyses documented in the plant LBD) comparable to the guideline exposures in applicable regulations are functions important to what are considered the most significant DBEs and shall be classified safety-related (see Sec. 4.3).

The safety analyses for the remaining DBEs shall be evaluated to identify any additional functions that are relied upon either to ensure the integrity of the RCPB or to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition. Any additional functions identified shall be classified safety-related (see Sec. 4.3).

3.3.3 Determine safety-related systems and structures.

The systems, including system-level structures (see Sec. 4.4), that are relied upon in the safety analyses of DBEs to provide or support the plant-level safety-related functions identified in Sec. 3.3.2 shall be determined. These systems and their corresponding plant-level safety-related functions shall be classified safety-related. Classification of systems should consider the presence of the most limiting single failure. Note that only those systems necessary and sufficient to accomplish any of the three basic safety-related functions need to be classified as safety-related. This may mean that not all sys-

tems capable of performing a given plant-level safety-related function need to be classified as safety-related. This also could mean that for some DBEs, not all plant-level safety-related functions are required.

3.3.4 Determine safety-related components.

The components (including component-level structures) that either are installed in the safety-related systems and are relied upon to provide or support the three basic safety-related functions or are installed in any system needed to satisfy safety-related interface requirements (e.g., isolation devices) shall be identified. These components shall be classified safety-related (see Sec. 4.5).¹⁹⁾ Note that only those components necessary and sufficient to accomplish one or more of the three basic safety-related functions need to be classified as safety-related. This may mean that not all components within a safety-related system need to be safety-related.

3.3.5 Determine safety-related parts.

The parts (including consumables) in the safety-related components that are relied upon to provide or support the three basic safety-related functions or satisfy safety-related interface requirements (i.e., where failures could prevent the satisfactory performance of a safety-related function) shall be classified safety-related (see Sec. 4.5). Note that only those parts necessary and sufficient to accomplish one or more of the three basic safety-related functions need to be classified as safety-related. This may mean that not all parts within a safety-related system or component need to be safety-related.

3.3.6 Determine non-safety-related with augmented requirements items.

Functions and associated items that have been evaluated and have not been classified as safety-related (Q) by Secs. 3.3.2 through 3.3.5, but are relied upon for a special event or are the subject of a significant licensing commitment, shall be classified as non-safety-related with augmented requirements (A) (see Sec. 4.6.2).

¹⁸⁾ Plant-level safety-related functions are defined by the licensee and can be similar to those developed for other regulatory purposes (such as development of EOPs).

¹⁹⁾ It is desirable to identify all Q component functions and A component functions to facilitate component qualification and parts classification.

3.3.7 Determine non-safety-related items.

Functions and associated items that have been evaluated and have not been classified safety-related (Q) by Secs. 3.3.2 through 3.3.5 or non-safety-related with augmented requirements (A) by Sec. 3.3.6 shall be classified non-safety-related (N).

4 Safety classification criteria

This section provides the criteria to be used with the methodology described in Sec. 3 to determine the safety classification of functions and items in a LWR nuclear power plant.²⁰⁾

4.1 General criteria

The process for determining an item's safety classification is based on the determination of DBEs and an item's safety-related function (or functions) with respect to those DBEs, as detailed in Secs. 3 and 4.2 through 4.6. This section describes general criteria that shall be used when determining an item's safety classification.

4.1.1 Functional analyses

Safety classification of items is based on the functions that the items are to perform in the safety analyses of DBEs and not on design criteria or operational requirements. Some items providing only non-safety-related functions might be required to meet selected design criteria normally applied to safety-related items. Such design criteria include Seismic Category I criteria; *ASME Boiler and Pressure Vessel Code* (B&PVC), Sec. III, criteria [23]; IEEE Class 1E criteria [24]; the general design criteria of 10 CFR 50, Appendix A; or NRC positions in regulatory guides that have been incorporated into the LBD. For example, some non-safety-related piping might be designed to Seismic Category I or B&PVC, Sec. III, Class 3, criteria. Neither these nor other design criteria are the bases for functionally classifying an item as safety-related but could be the basis for classifying an item as non-safety-related with augmented requirements.

Similarly, some items providing only non-safety-related functions might be subject to

operational requirements such as technical specifications, EOPs, or surveillance procedures. For example, non-safety-related valves might have surveillance requirements set forth in the technical specifications, or their operation might be specified in EOPs. Operational requirements such as these, or any others, are not bases for functionally classifying an item as safety-related.

4.1.2 Licensing requirements

For the purposes of this standard, only the functional role an item plays in meeting the regulatory definition of safety-related is to be used to classify the item. However, it is recognized that classifications of some SSCs have historically been based on licensing commitments. Consequently, in the determination of the safety classification of an item, licensing guidance or other guidance (e.g., industry standards) shall be deemed a requirement when a commitment to comply with the guidance is specified in the plant LBD.

4.1.3 Pre-design basis event availability

Items relied upon in the safety analysis of DBEs to support one or more of the three basic safety-related functions shall be classified as safety-related. For example, items required to provide emergency coolant to the reactor vessel after a LOCA shall be classified safety-related. However, items required only to ensure the pre-DBE availability of such items may be classified non-safety-related. For example, if operation of a pump that keeps an ECCS line full is not relied upon during or following a LOCA, then its pumping function may be classified non-safety-related. (Other criteria might require other functions of the pump, such as pressure integrity, to be safety-related.)

4.1.4 Operability of items

Items shall be considered to be in an operable condition (e.g., filled, vented, and with valves lined up in compliance with the operating procedures and technical specifications) in the evaluation to determine safety classification. Degraded system availability due to maintenance, repair, or testing is addressed by technical specifications and shall not be addressed in the safety classification evaluation.

²⁰⁾ Examples of the application of these criteria are provided in Appendix C.

4.1.5 Installation and maintenance

All items (both safety-related and non-safety-related) shall be assumed to be present, properly installed, and properly maintained in the evaluation to determine safety classification. The consequences of missing, incorrect, or improperly installed or maintained items shall not be addressed in the safety classification evaluation.

4.1.6 Impairment of safety-related functions

An item that does not perform or support a safety-related function, but whose failure could prevent the satisfactory accomplishment of a safety-related function, shall be classified safety-related unless otherwise justified (see Sec. 4.5.2).

4.1.7 Failures

Only failures occurring during or following DBEs, including single failures in accordance with the single-failure criterion (see Sec. 4.1.9), need to be evaluated in determining safety classification. Failures prior to a DBE shall not be addressed in the safety classification evaluation.²¹⁾ Nondetectable failures²²⁾ shall be assumed to occur concurrently with the DBE.

4.1.8 Verified design

Safety classification is the process of determining and independently verifying whether an item is safety-related. Safety classification shall be based on the assumption that all other design characteristics have been previously verified. When the safety classification of an item is being determined, all other design characteristics of all items shall be assumed to be correct. The possibility of design errors shall not be evaluated in determining safety classification.

4.1.9 Single-failure criterion

The application of the single-failure criterion shall assume (a) the initiating cause of the DBE and (b) an additional single failure in an item providing the safety-related functions or item connected to that safety-related item.

4.1.10 Interfaces

Items that do not perform a safety-related function and are connected to or interface with safety-related items may be classified non-safety-related if (a) the safety-related items are isolated by safety-related isolation devices (e.g., optical isolators or automatic valves) from the effects of failures in the non-safety-related items or (b) an evaluation justifies an assertion that a failure in the non-safety-related items (including a potential common-cause failure) will not prevent accomplishment of a safety-related function of the safety-related items.

4.1.11 Satisfying classification requirements with multiple items

A plant design function may be provided by two or more items to separately meet different requirements, provided that the need for those requirements will not occur simultaneously during or following any DBE. For example, two ultimate heat sinks could be provided: one tornado-proof, but not seismically qualified, and the other seismically qualified, but not tornado-proof. Because the simultaneous occurrence of a tornado and a seismic event is not a DBE, these requirements may be divided between two items, each of which shall be classified safety-related for the DBE for which they are required.

4.1.12 Mission time of safety-related functions

The time following a DBE that a function is required to be safety-related may be limited on a case-by-case basis. For example, on-site diesel fuel oil supply may be a safety-related function; however, the resupply of this diesel fuel oil is not a safety-related function. The urgency of the function and the availability of items and resources to perform the function shall be the basis for such determinations.

4.1.13 Documentation

The safety classification evaluation of an item, e.g., results of classification and regulatory basis for the classification (events, functions, etc.), shall be documented in accordance with the

²¹⁾ The availability of an item and associated actions due to failures prior to a DBE are controlled by technical specifications.

²²⁾ See Sec. 2.2 for definition of nondetectable failures.

quality assurance requirements of 10 CFR 50, Appendix B.

4.2 Determination of design basis events

The LBD contains analyses of events, both DBEs and special events, upon which the design of the plant has been or will be based and licensed. The DBEs identified shall include the subset of DBEs that could result in potential off-site exposures (as described in the safety analyses of DBEs summarized in the plant LBD) comparable to the guideline exposures in applicable regulations (assuming that all mitigation features function as described in the analyses).

Special events (such as ATWS, SBO, and fire) typically are initiated by common-cause or multiple-cause failures. While special events are part of the licensing basis of the plant, they are not DBEs, and non-safety-related functions or items may be credited to prevent or mitigate their consequences.

4.3 Determination of safety-related functions

4.3.1 General functional criteria

Plant-level safety-related functions are those functions that are relied upon in DBE analyses during or following a DBE to ensure that any of the three basic safety-related functions can be accomplished. Any function whose failure prevents the accomplishment of one or more of the three basic safety-related functions shall be classified safety-related unless otherwise justified (see Sec. 4.5).

Plant-level safety-related functions and safety-related systems shall be determined through review of the safety analyses of the DBEs described in the LBD per Secs. 3.3.2 and 3.3.3.²³⁾ Sections 4.3.1.1 through 4.3.1.4 clarify the wording of the three basic safety-related functions.

4.3.1.1 Reactor Coolant Pressure Boundary Integrity

Ensuring the integrity of the RCPB shall be a safety-related function; however, the RCPB is

not leak tight. The RCPB integrity is maintained if the leakage from it is within the normal reactor coolant makeup capability.²⁴⁾ All portions of the RCPB whose failure would not affect the integrity of the RCPB may be classified non-safety-related unless other functional requirements require the item to be classified safety-related. For example, if failure of a reactor vessel head closure O-ring seal could not cause a loss of coolant in excess of the normal makeup capacity, then the seal may be classified non-safety-related.

4.3.1.2 Acceptance limits for “Shut Down the Reactor”

Ensuring the capability to shut down the reactor shall be a safety-related function. Not all equipment that is capable of reactor shutdown has to be safety-related. The acceptance limits for the function of shutting down the reactor shall be the guideline exposures in applicable regulations. For example, the reactor trip (scram) function is classified safety-related, but other reactivity control functions (e.g., reactor core coolant flow control and movement of control rods) that assist in achieving a reactor shutdown may be classified as non-safety-related. Other plant items that are relied upon to satisfy only design basis transient limits (e.g., turbine controller and feedwater controller) need not be classified safety-related.

4.3.1.3 Potential off-site exposures

Potential off-site exposures are the calculated exposures specified in the safety analyses of DBEs as summarized in the plant LBD. These exposures are based on use of plant systems to mitigate a DBE in accordance with the assumptions in the safety analysis of the DBE.

4.3.1.4 Comparable off-site exposures

The term “comparable” in the regulatory definition of safety-related is subjective. To ensure consistent interpretation, “comparable” is defined in this standard as $\geq 10\%$. This value is chosen because (a) it is consistent with NRC

²³⁾ See Sec. B.6 in Appendix B for a discussion of active and passive functions.

²⁴⁾ This statement does not alter, and is compatible with, the regulatory definition of the RCPB (see 10 CFR 50.2 and 10 CFR 50.55a). In currently licensed BWR plants, the normal reactor coolant makeup capability, in this context, is the reactor core isolation cooling system flow, and in pressurized water reactor plants, it is the charging system flow.

guidance²⁵); (b) philosophically, within an order of magnitude it is comparable while more than an order of magnitude is not; and (c) it yields results consistent with past and current industry and NRC practice.

The NRC guidance states that the radiological consequences of DBE should “not exceed,” be “well within,” or be a “small fraction” of the guideline exposures in applicable regulations. This NRC guidance defines the terms “not exceed” to mean $\leq 100\%$ of guideline exposures, “well within” to mean $< 25\%$ of guideline exposures, and “small fraction of” to mean $< 10\%$ of guideline exposures. “Not exceed” and “well within” are “comparable”; however, “small fraction” is not “comparable.” This amplification of the safety-related definition recognizes that there is a threshold value for off-site exposures that defines the boundary between safety-related and non-safety-related.

4.3.2 Specific functional criteria

Besides the plant-level safety-related functions identified in Sec. 3.3.2, specific supporting functions shall be identified and classified safety-related for specific DBEs, for example,

- (1) maintaining an environment to ensure that a safety-related item can perform its safety-related function;
- (2) ensuring the habitability in the control room during or following a DBE;
- (3) providing power, such as from an electrical power source or from compressed gas, to ensure that a safety-related item can perform its safety-related function;
- (4) providing a floodable volume to maintain adequate water level in the spent-fuel storage pool to prevent gross failure of the spent fuel due to high temperatures;
- (5) providing functional support, such as cooling water or lubricating oil, to ensure that a safety-related item can perform its safety-related function;
- (6) providing effluent control provisions outside the primary containment to prevent the release of radioactive material that could result in potential off-site exposures compara-

ble to guideline exposures in applicable regulations; or

- (7) providing structural support for a safety-related item to ensure that the item can perform its safety-related function.

4.4 Determination of safety-related systems and structures

Systems, including system-level structures (i.e., those structures that have, or could have, system-level plant-unique identification codes assigned), shall be classified using the criteria in Secs. 4.4.1 through 4.4.4.

4.4.1 Design functions

The portions of a system that are relied upon to perform any of the three safety-related functions shall be classified as safety-related for those functions.

4.4.2 Modes of operation

The classification of each system shall be based on the evaluation of all its modes of operation. Potential functional requirements and lineups of a system for potential plant operating modes shall be evaluated when the system is being classified.

4.4.3 Licensing commitments

Some systems or portions of systems that are not required to provide a safety-related function may have licensing commitments that apply criteria typically applied to safety-related items. These systems or portions of systems shall be classified as non-safety-related with augmented requirements.

4.4.4 Functional boundaries

Functional boundaries for each safety-related function of a system should be determined with the aid of the criteria of Sec. 4.5 to facilitate the classification of components and parts. The system functional boundaries should be identified using piping and instrumentation drawings (P&IDs). The electrical drawings, electrical schematics, single-line diagrams, and instrument loop diagrams should be used to identify electrical items and functional boundaries that cannot be identified on the P&IDs.

²⁵ See Regulatory Guide 1.70 [25] and NUREG-0800, Chapter 15 [26].

4.5 Determination of safety-related components and parts

Components, including component-level structures (i.e., those structures that have, or could have, component-level plant-unique identification codes assigned), and parts (including consumables) of each system shall be classified through the use of the following criteria. First, the component or part should be located on the pertinent system or detail drawing, and the functions that have been identified for the system should be reviewed (see Sec. 4.4). The component or part may then be classified through the use of (a) pertinent system documentation; (b) the DBE, safety-related functions, and safety-related systems determined through application of the criteria of Secs. 4.2, 4.3, and 4.4; and (c) the criteria discussed in Secs. 4.5.1 through 4.5.6.

4.5.1 General criteria for components and parts

Consumables (e.g., hydraulic fluid, chemicals, gaskets, and lock wire) shall be classified in accordance with the safety classification criteria for associated parts.

The classification of a component or part shall be based upon its function. If the component or part is relied upon for the satisfactory performance of a safety-related function, the component or part shall be classified safety-related.

A component or part that is relied upon explicitly or implicitly in an analysis of a DBE to provide a safety-related function shall be classified safety-related.

A component or part that is relied upon to satisfy a safety-related interface requirement shall be classified safety-related.

A component or part that is relied upon to initiate, control, or maintain a safety-related function shall be classified safety-related.

A component or part that is relied upon during or following a DBE to maintain an environment to ensure that a safety-related item can perform its safety-related function shall be classified safety-related.

A component or part that is relied upon to prevent the release of radioactive material that could result in potential off-site exposures comparable to guideline exposures in applicable regulations, during or following a DBE, shall be classified safety-related.

A component or part that does not perform or support a safety-related function, but whose failure could prevent the satisfactory performance of a safety-related function of another item, shall be classified safety-related unless otherwise justified.

If the function of a component or part is not safety-related, and its failure could not prevent the satisfactory performance of a safety-related function, the component or part shall be classified non-safety-related with augmented requirements or non-safety-related.

A safety-related component might contain parts that are safety-related, non-safety-related with augmented requirements, or non-safety-related.

4.5.2 Credibility of failures of components and parts

When evaluating failures of a component or part that does not perform or support a safety-related function, any potential failure of the component or part shall be deemed credible unless justified otherwise in accordance with an established basis. Such justification may be based upon the following:

- (1) The failure is not physically possible for the functional or environmental conditions that would exist during or following a DBE;
- (2) The component or part has been qualified to resist the failure. For example, for a component or part that is Seismic Category I or II, failures due to a SSE do not need to be assessed. For a component or part that is environmentally qualified, failures due to environmental conditions for which it has been qualified do not need to be assessed;
- (3) A probability evaluation shows that the frequency of occurrence for the DBE sequence including the failure is $<10^{-7}$ /year or $<10^{-6}$ /year if when combined with reasonable qualitative arguments, the realistic frequency can be shown to be lower²⁶⁾;

²⁶⁾ See NUREG-0800, Sec. 2.2.3.