



Nuclear Safety Criteria and Design Process for Sodium Fast Reactor Nuclear Power Plants

An American National Standard

Published by the
American Nuclear Society
555 N. Kensington Ave
La Grange Park, IL 60526



ANSI/ANS-54.1-2020

**American National Standard
Nuclear Safety Criteria and
Design Process for Sodium Fast
Reactor Nuclear Power Plants**

Secretariat
American Nuclear Society

Prepared by the
**American Nuclear Society
Standards Committee
Working Group ANS-54.1**

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

Approved March 23, 2020
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 It may be reaffirmed, revised, or withdrawn at any time in accordance with established procedures.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. Inquiries about requirements, recommendations, and/or permissive statements (i.e., “shall,” “should,” and “may,” respectively) should be sent to the Society headquarters, ATTN: Standards or to standards@ans.org. Action will be taken to provide appropriate response in accordance with established procedures that ensure consensus.

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



This document is copyright protected.

Copyright © 2020 by American Nuclear Society. All rights reserved.

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 a specific design, operation, facility, or other unique situation only and therefore is not intended for generic application. An inquiry submittal form is available at <https://ans.org/standards/docs/inquiry-submittal-form.pdf>.

Responses to inquiries on standards are published in ANS’s magazine, *Nuclear News*, and are available publicly at www.ans.org or by contacting standards@ans.org.

Inquiry Format

Inquiry requests shall 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 purpose(s) 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
555 N. Kensington Avenue
La Grange Park, IL 60526

or standards@ans.org

Foreword

(This foreword does not contain any requirements of American National Standard “Nuclear Safety Criteria and Design Process for Sodium Fast Reactor Nuclear Power Plants,” ANSI/ANS-54.1-2020, but is included for informational purposes.)

The overarching purpose of this standard is to define criteria to be satisfied for providing reasonable assurance that sodium fast reactor (SFR) nuclear power plants are designed so that they can be operated with adequate protection of the public health and safety and acceptable environmental impacts as defined by the national regulator. This purpose is achieved through the identification of applicable safety requirements from the U.S. Nuclear Regulatory Commission (NRC), industrial codes and standards, and other published guidance and professional engineering practices. It is also the purpose of this standard to define requirements for the acceptable use of probabilistic risk information in support of the design process (i.e., risk-informed design criteria). These latter requirements using risk information are optional in nature.

This standard defines safety objectives; sodium fast reactor design criteria (SFRDCs); selection criteria for licensing-basis events; and criteria for the classification of structures, systems, and components (SSCs) that can be used by designers and regulators of SFR nuclear power plants. It is intended to provide the necessary guidance to the designer in order to “bridge” the existing light water reactor (LWR)–focused general design criteria (GDCs) contained in Appendix A to 10 CFR 50 and other regulatory requirements to the development of their respective principal design criteria, while retaining the underlying safety principles of the GDCs. This standard builds on a U.S. Department of Energy (DOE)–sponsored Idaho National Laboratory report and a new NRC regulatory guide, which are discussed below.

This standard is intended to establish the details in support of the approach for licensing and regulation of SFR nuclear power plants. As with other standards that have been developed by standards development organizations, it is anticipated that this standard will be reviewed by the national nuclear regulator who will, in turn, use it as input for issuance of regulatory guidance specific for this class of reactors.

This standard applies to all SFR nuclear power plants, irrespective of level of power production and energy end use. It also applies to configurations in which there are one or more reactor units (modules) on a site. It is intended to apply to all fuel types. The heat transport system is not restricted to a particular configuration, and thus, this standard applies to loop, pool, hybrid, or other arrangements. This standard also pertains to on-site storage of spent fuel prior to its removal for recycling or long-term storage.

This standard does not directly address plant security design requirements or criteria. However, it does note the importance of integrating security into the design of the reactor to alleviate the possibility of conflicts between design and security requirements. The NRC has recently published draft security design considerations for advanced reactors. Once finalized by the NRC, these should be considered in conjunction with the safety design criteria and requirements contained in this standard. Security design requirements, including design-basis threats, are design elements to be brought into the plant design process to address licensing requirements of the national nuclear regulator. In general, both deterministic and risk-informed approaches may be considered in the plant security design process as provided by the national nuclear regulator. It is anticipated that various safety features and inherent safety characteristics together with the use of the risk-informed nuclear safety process described herein will effectively support plant security design.

The risk-informed process embedded in the requirements of this standard is voluntary rather than mandatory. It represents a departure from the approach used by professional

communities familiar with traditional, deterministic LWR design processes. These communities include nuclear plant architect/engineer, nuclear licensing, and risk assessment professionals. This process presents an opportunity to extend traditional use of probabilistic risk assessment as applied to LWRs to SFRs and to incorporate risk insights early in the design process.

In addition, to designers, regulators, and the risk community, this standard provides a tool for plant operators who use design processes to maintain licensed plant designs. Some uses of this standard, such as identifying SSCs that require special treatment, apply beyond initial plant design, procurement, and construction into operations. Use of this standard for that purpose also allows plant owner/operators to specify special treatments over the life of the plant for procurement, application, testing, and maintenance commensurate with risk. This standard documents an established process that nuclear design organizations can use to develop nuclear safety designs. It is anticipated that sponsors of SFR designs will develop further specific designs with design-dependent standards that integrate these risk-informed characteristics.

An earlier version of this standard, ANS-54.1-1989, “General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant,” was issued in 1989 and withdrawn in 1999 due to waning interest in SFRs, although two LWR counterparts, ANS-51.1/N18.2-1971, “Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants,” and ANS-52.1-1978, “Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants,” were completed and approved much earlier (1971 and 1978, respectively). Those standards were subsequently updated in 1983, adding new risk-based classifications. However, along with the earlier ANS-54.1 standard, all remained essentially deterministic compilations of the state-of-the-art design from that era. Revived interest in SFRs spurred by the DOE and several industrial organizations, including the development in the industry of certain small modular reactor designs, has motivated the development of this new standard. Other, more detailed implementation standards are envisioned for detailed designs.

The working group that has prepared this standard was formed in 2010, and a reasonably complete draft was ready in 2014, but at that time the working group decided to put its completion in abeyance until an important effort being undertaken by the DOE and then by the NRC to develop new, broad recommendations for design criteria had reached fruition. When the DOE effort produced a useful report in late 2014, INL/EXT-14-31179, “Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors,” followed by the NRC’s publication of its Regulatory Guide 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors,” the working group resumed its work and completed the draft standard in 2018, beginning the American Nuclear Society’s committee approval process. As noted, the SFRDCs in this standard build upon the above-cited DOE and NRC work.

Use of this standard does not supersede the responsibility to review and apply the top-level safety criteria of the regulatory authorities in the country in which the user plans to license, build, and operate a SFR. This standard may also be used to support the preparation of a SFR safety analysis report for the purpose of licensing. When used for licensing, this standard does not provide the only basis for establishing the safety and design criteria. The designer must also assess the applicability of the existing body of technical licensing requirements and guidance for nuclear plant licensing in the particular country of application. In this regard, the designer determines the applicability, partial applicability, or nonapplicability of these licensing requirements. The designer may also use this standard and other supporting standards to determine what additional licensing technical requirements are required for important technical

design and safety aspects that are not addressed by the existing body of technical licensing requirements and guidance.

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

This standard incorporates the concepts of generating risk-informed insights and performance-based requirements.

This standard was prepared by the ANS-54.1 Working Group of the American Nuclear Society. The following individuals contributed to this standard:

G. F. Flanagan (Chair), *Oak Ridge National Laboratory*
R. J. Budnitz (Vice Chair), *Lawrence Berkeley National Laboratory (retired)*

R. A. Bari, *Brookhaven National Laboratory*
K. A. El-Sheikh, *The Cameron Group, Inc.*
P. Gaillard, *TerraPower*
M. E. Garrett, *Individual*
C. W. Grandy, *Argonne National Laboratory*
A. Greci, *Salt River Project*
N. P. Kadambi, *Individual*
T. Kevern, *U.S. Nuclear Regulatory Commission*
T. L. King, *Individual*
E. P. Loewen, *GE Hitachi Nuclear Energy*
C. Lobscheid, *Advent Engineering*
I. K. Madni, *U.S. Nuclear Regulatory Commission*
H. Matsumiya, *Toshiba*
J. Mazza, *U.S. Nuclear Regulatory Commission*
A. Miller, *Areva Enrichment Service*
Y. Okano, *Japan Atomic Energy Agency*
M. Vidard, *Électricité de France*
R. A. Wigeland, *Idaho National Laboratory*

The Advanced Initiatives Subcommittee had the following membership at the time of its approval of this standard:

B. B. Bevard (Chair), *Oak Ridge National Laboratory*
A. Afzali, *Southern Company*
J. K. August, *Southern Company*
E. D. Blandford, *Kairos Power LLC*
G. F. Flanagan, *Oak Ridge National Laboratory*
D. E. Holcomb, *Oak Ridge National Laboratory*
M. A. Linn, *Oak Ridge National Laboratory*
D. L. Moses, *Individual*
R. D. Sachs, *Individual*

The Research and Advanced Reactors Consensus Committee had the following membership at the time of its approval of this standard:

G. F. Flanagan (Chair), *Oak Ridge National Laboratory*
B. B. Bevard (Vice Chair), *Oak Ridge National Laboratory*
T. H. Newton (Vice Chair), *National Institute of Standards & Technology*

A. Adams, Jr., *U.S. Nuclear Regulatory Commission*
A. Afzali, *Southern Company*
J. K. August, *Southern Company*
E. D. Blandford, *Kairos Power LLC*
R. E. Carter, *Individual*
L. P. Foyto, *University of Missouri*
A. Greci, *Salt River Project*
B. K. Grimes, *Individual*
D. E. Holcomb (Observer), *Oak Ridge National Laboratory*
D. R. Lawson, *U.S. Department of Energy*
M. A. Linn, *Oak Ridge National Laboratory*
J. Mazza, *U.S. Nuclear Regulatory Commission*
M. J. Memmott, *Brigham Young University*
D. S. O'Kelly, *Idaho National Laboratory*
M. W. Peres, *Fluor Enterprises Inc.*
S. R. Reese, *Oregon State University*
R. S. Turk, *Individual*
A. R. Veca, *General Atomics*

Contents

Section	Page
1 Introduction, scope, purpose	1
1.1 Purpose of standard.....	1
1.2 Scope and history	1
1.3 Summary of overall approach used in this standard	2
1.3.1 Principal design criteria, their role, and their relationship to the SFRDCs in Sec. 3.....	4
1.4 PRA scope and capability	4
1.4.1 PRA scope	4
1.4.2 Minimum requirement for the PRA.....	5
2 Definitions and acronyms.....	5
2.1 Shall, should, and may	5
2.2 Definitions	6
2.3 Acronyms.....	7
3 Sodium fast reactor design criteria	8
3.1 Overall requirements.....	8
3.1.1 Criterion 1 – quality standards and records.....	8
3.1.2 Criterion 2 – design bases for protection against natural phenomena.....	8
3.1.3 Criterion 3 – fire protection.....	8
3.1.4 Criterion 4 – environmental and dynamic effects design bases.....	8
3.1.5 Criterion 5 – sharing of structures, systems, and components.....	9
3.2 Multiple barriers.....	9
3.2.1 Criterion 10 – reactor design	9
3.2.1.1 Criterion 10 – recommended supplemental reactor design criteria.....	9
3.2.2 Criterion 11 – reactor inherent protection	9
3.2.3 Criterion 12 – suppression of reactor power oscillations	9
3.2.4 Criterion 13 – instrumentation and control	10
3.2.5 Criterion 14 – primary coolant boundary	10
3.2.6 Criterion 15 – primary coolant system design	10
3.2.7 Criterion 16 – containment design.....	10
3.2.8 Criterion 17 – electric power systems	10
3.2.9 Criterion 18 – inspection and testing of electric power systems.....	10
3.2.10 Criterion 19 – control room.....	11
3.3 Reactivity control.....	11
3.3.1 Criterion 20 – protection system functions	11
3.3.2 Criterion 21 – protection system reliability and testability	11
3.3.3 Criterion 22 – protection system independence.....	11
3.3.4 Criterion 23 – protection system failure modes.....	11
3.3.5 Criterion 24 – separation of protection and control systems.....	12
3.3.6 Criterion 25 – protection system requirements for reactivity control malfunctions.....	12
3.3.7 Criterion 26 – reactivity control systems.....	12
3.3.8 Criterion 28 – reactivity limits	12
3.3.9 Criterion 29 – protection against anticipated operational occurrences.....	12
3.4 Fluid systems	12
3.4.1 Criterion 30 – quality of primary coolant boundary	12

3.4.2	Criterion 31 – fracture prevention of primary coolant boundary.....	13
3.4.3	Criterion 32 – inspection of primary coolant boundary	13
3.4.4	Criterion 33 – primary coolant inventory maintenance.....	13
3.4.5	Criterion 34 – residual heat removal.....	13
3.4.6	Criterion 35 – emergency core cooling	13
3.4.7	Criterion 36 – inspection of emergency core cooling system.....	13
3.4.7.1	Criterion 36 – recommended supplemental design criteria.....	14
3.4.8	Criterion 37 – testing of emergency core cooling system	14
3.4.8.1	Criterion 37 – recommended supplemental design criteria.....	14
3.4.9	Criterion 38 – containment heat removal.....	14
3.4.10	Criterion 39 – inspection of containment heat removal system	14
3.4.11	Criterion 40 – testing of containment heat removal system	14
3.4.11.1	Criterion 40 – recommended supplemental design criteria.....	14
3.4.12	Criterion 41 – containment atmosphere cleanup.....	14
3.4.13	Criterion 42 – inspection of containment atmosphere cleanup systems.....	15
3.4.14	Criterion 43 – testing of containment atmosphere cleanup systems.....	15
3.4.15	Criterion 44 – structural and equipment cooling	15
3.4.16	Criterion 45 – inspection of structural and equipment cooling systems.....	15
3.4.17	Criterion 46 – testing of structural and equipment cooling systems.....	15
3.4.17.1	Criterion 46 – recommended supplemental design criteria.....	15
3.5	Reactor containment.....	15
3.5.1	Criterion 50 – containment design basis.....	15
3.5.2	Criterion 51 – fracture prevention of containment boundary.....	16
3.5.3	Criterion 52 – capability for containment leakage rate testing	16
3.5.4	Criterion 53 – provisions for containment testing and inspection	16
3.5.5	Criterion 54 – piping systems penetrating containment	16
3.5.6	Criterion 55 – primary coolant boundary penetrating containment.....	16
3.5.7	Criterion 56 – containment isolation	17
3.5.8	Criterion 57 – closed system isolation valves	17
3.6	Fuel and radioactivity control.....	17
3.6.1	Criterion 60 – control of releases of radioactive materials to the environment.....	17
3.6.2	Criterion 61 – fuel storage and handling and radioactivity control	17
3.6.2.1	Criterion 61 – recommended supplemental design criteria.....	18
3.6.3	Criterion 62 – prevention of criticality in fuel storage and handling.....	18
3.6.4	Criterion 63 – monitoring fuel and waste storage.....	18
3.6.5	Criterion 64 – monitoring radioactivity releases.....	18
3.7	Additional SFRDCs	18
3.7.1	Criterion 70 – intermediate coolant systems.....	18
3.7.2	Criterion 71 – primary coolant and cover gas purity control	18
3.7.3	Criterion 72 – sodium heating systems.....	18
3.7.3.1	Criterion 72 – recommended supplemental design criteria.....	18
3.7.4	Criterion 73 – sodium leakage detection and reaction prevention and mitigation.....	19
3.7.4.1	Criterion 73 – recommended supplemental design criteria.....	19
3.7.5	Criterion 74 – sodium–water reaction prevention/mitigation	19
3.7.6	Criterion 75 – quality of the intermediate coolant boundary	19
3.7.7	Criterion 76 – fracture prevention of the intermediate coolant boundary	19
3.7.8	Criterion 77 – inspection of the intermediate coolant boundary.....	19
3.7.9	Criterion 78 – primary coolant system interfaces.....	19
3.7.10	Criterion 79 – cover gas inventory maintenance	20

3.8	Other design criteria.....	20
3.8.1	Criterion 80 – liquid sodium receiving, storage, and processing systems.....	20
3.8.2	Criterion 90 – aging management.....	20
3.8.3	Criterion 91 – human factors.....	20
3.8.4	Criterion 92 – physical security/safeguards	21
3.8.5	Criterion 93 – emergency access	21
3.9	Beyond-design-basis accidents.....	21
3.9.1	Criterion 100 – station blackout.....	21
3.9.2	Criterion 101 – anticipated transients without scram	21
3.9.3	Criterion 102 – aircraft impact.....	22
3.9.4	Criterion 103 – severe accidents	22
3.10	Using the PDCs	22
4	Risk-informing the design process.....	22
4.1	Introduction.....	22
4.1.1	Application and purpose	22
4.1.2	Scope.....	23
4.2	Process for developing risk-informed, performance-based PDCs	24
4.3	Risk-informed acceptance criteria.....	26
4.3.1	Accident prevention	26
4.3.2	Accident mitigation.....	26
4.3.3	Risk management.....	27
4.4	Licensing-basis events: Identification of LBEs, selection criteria, and application in the design process	27
4.4.1	Introduction	27
4.4.2	Use of LBEs in developing a safety case	28
4.4.3	How the PRA is to be used.....	28
4.4.4	The fundamental principle in using LBEs.....	29
4.4.5	Special treatment of SSCs	30
4.4.6	Identification of the set of LBE sequences using the PRA: Potential pitfalls and difficulties	30
4.4.7	PRA uncertainties	31
4.4.8	Step-by-step approach for LBE selection.....	31
4.4.9	Acceptance criteria.....	33
4.5	Identify dominant accident sequences and SSCs that require special treatment.....	33
4.5.1	Identify dominant accident sequences	33
4.5.2	Identify SSCs that require special treatment.....	34
5	References	34

Tables

Table 1	Proposed frequency categories for LBE sequences.....	26
Table 2	Proposed acceptance criteria for LBE sequences	33

Figures

Figure 1	High-level step-by-step process for risk-informing the design.....	23
Figure 2	Risk-informing the design process.....	25