



ANS

Auxiliary Feedwater System for Pressurized Water Reactors

An American National Standard

ANSI/ANS-51.10-2020

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



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**American National Standard
Auxiliary Feedwater
System for Pressurized
Water Reactors**

Secretariat
American Nuclear Society

Prepared by the
**American Nuclear Society
Standards Committee
Working Group ANS-51.10**

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

Approved October 23, 2020
by the
American National Standards Institute, Inc.

American National Standard

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Published by

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



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Foreword

(This foreword does not contain any requirements of American National Standard “Auxiliary Feedwater System for Pressurized Water Reactors,” ANSI/ANS-51.10-2020, but is included for informational purposes.)

This standard is applicable to pressurized light water reactor nuclear power plants using auxiliary feedwater for emergency applications. Plants that rely on passive safety systems independent of the alternating-current (ac) power system are not considered in the scope of this standard.

ANS-51.10 was originally issued in 1979 and has been extensively updated to reflect current regulatory directives, industry practice and experience, and available design guidance.

Among the major changes incorporated in the 1991 revision of ANS-51.10 was station blackout (i.e., the loss of all ac power sources). A requirement was included that the system be capable of operating for a plant-specific duration with the loss of all ac power sources. The rationale for this requirement and the method of determining the plant-specific duration are identified in Title 10, *Code of Federal Regulations*, Part 50, Sec. 50.63, “Loss of All Alternating Current Power.” In the 1979 edition of this standard, a generic duration of 2 hours was identified, and no rationale was included.

Major revision for ANSI/ANS-51.10-2020 includes additional clarification on defense-in-depth and diversity of power sources as well as updated references and probabilistic risk assessment language where appropriate.

This standard might reference documents and other standards that have been superseded or withdrawn at the time the standard is applied.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 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.

The membership of the ANS-51.10 Working Group of the American Nuclear Society during the revision of this standard was as follows:

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Auxiliary Feedwater System for Pressurized Water Reactors

1 Scope, purpose, and application

1.1 Scope

This standard¹⁾ sets forth the safety-related functional requirements, performance requirements, design criteria, design requirements for testing and maintenance, and interfaces for the safety-related portion of the auxiliary feedwater system (AFS) of pressurized water reactor (PWR) plants.

This standard is written for new facilities that rely on an auxiliary (emergency) feedwater system for a safety-related function.

1.2 Purpose

This standard is written for new facilities. This standard reflects existing regulations, design guidance, and operational experience.

The applicability of this standard to a specific operating plant must be determined on a case-by-case basis to be in compliance with the plant's licensing bases. The criteria and guidance provided in this standard may be different from and inconsistent with the licensed design bases of an operating plant.

1.3 Limits of application

This standard applies to the safety-related AFS. This standard also discusses relevant requirements associated with the AFS instrumentation and controls, source of power, water supply, and support system interfaces.

2 Acronyms and definitions

2.1 Acronyms

ac:	alternating-current
AFS:	auxiliary feedwater system
ANS:	American Nuclear Society
ANSI:	American National Standards Institute
ASME:	American Society of Mechanical Engineers
CFR:	<i>Code of Federal Regulations</i>
dc:	direct-current
ESF:	engineered safety feature

¹⁾ The current standard, ANSI/ANS-51.10-2020, is hereinafter referred to as "this standard."

LOCA:	loss-of-coolant accident
NRC:	U.S. Nuclear Regulatory Commission
PWR:	pressurized water reactor
RCS:	reactor coolant system
RHR:	residual heat removal
SSC(s):	structure(s), system(s), or component(s)

2.2 Shall, should, and may

shall, should, and may: The word “shall” is used to denote a requirement; the word “should” is used to denote a recommendation; and the word “may” is used to denote permission, neither a requirement nor a recommendation.

2.3 Definitions

active failure: A malfunction of a component that prevents mechanical movement or change of state required to accomplish the function of the component on demand. Examples of active failures include the failure of a valve or check valve to move to its correct position or the failure of a pump, fan, or diesel generator to start.

Spurious action of a powered component originating within its actuation or control system is regarded as an active failure unless specific design features or operating restrictions preclude such spurious action. An example is the unintended energization of a powered valve to open or close.

engineered safety feature (ESF): A nuclear safety–related structure, system, or component (SSC) that serves to control and limit the consequences of releases of energy or radioactivity if an event were to occur to the extent that worker and public health and safety might be impaired if these energy or radioactivity releases were not additionally restrained.

hot standby: In the PWR, the condition in which the reactor is subcritical and the reactor coolant system (RCS) average temperature is above the required temperature to permit operation of the low-pressure residual heat removal (RHR) system, consistent with technical specification operational limits.

minimum delivered flow: The amount of flow that must be delivered to the intact steam generator(s).

minimum recirculation flow: The amount of flow that must be provided at all times that the pump is operating to protect the pump from overheating and accelerated aging.

nuclear safety function: Any function that is necessary to ensure

- (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 10 CFR 100 [1].²⁾

²⁾ Numbers in brackets refer to corresponding numbers in Sec. 7, “References.”

operator error: In the context of the single-failure criterion, a single incorrect or omitted action by a human operator attempting to perform a safety-related manipulation in response to an initiating occurrence.

passive failure: The blockage of a process flow path or failure of a component to maintain its structural integrity or stability such that it cannot provide its intended nuclear safety function upon demand.

seismic Category I: The category of SSCs that are designed to perform at least one function during or after a safe shutdown earthquake.

single failure: A random failure (e.g., single-component failure or operator error) and its consequential effects, in addition to an initiating occurrence, which result in the loss of capability of a component to perform its intended function.

Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) 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 nuclear safety function(s).

Note: The term refers to a failure that is assumed to occur in addition to an initiating occurrence for the purposes of safety class system design analysis as described in

- (1) 10 CFR 50, “Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants” [2];
- (2) IEEE 379-2014, “Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems” [3];
- (3) ANSI/ANS-58.9-2002 (R2020), “Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems” [4].³⁾

3 System functions

The AFS shall provide flow to the steam generators in the event that the main feedwater system is not available. The AFS shall operate for all Plant Condition 2, 3, 4, and 5 events where main feedwater supplies are not available and the steam generators are used to mitigate the event. Plant Condition 1, 2, 3, 4, and 5 events are defined in ANS-51.1-1983 (R1988) (withdrawn) [6].

The AFS shall perform its main safety function as stated above without overflowing a steam generator to prevent steam generator overfill or in response to a steam generator tube rupture.

The AFS may be designed to also perform functions in support of normal plant start-up and shutdown unless the plant design does not include normal plant functions for this system.

4 System definition

4.1 Description of AFS

This standard defines the extent and boundaries of the AFS for PWRs and interfaces with related systems or subsystems.⁴⁾

³⁾ For additional guidance, see U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.53, Rev. 2 [5].

⁴⁾ Refer to Appendix A.

5 System performance requirements

5.1 Performance parameters

The function of the AFS is to deliver required flow to the operable steam generator(s) under the plant conditions identified in Sec. 3. The system shall be designed to ensure that the required flow can be delivered when needed and for as long as the system function is required (see Secs. 5.1.1 and 5.1.2). This includes selection of adequate pump head flow characteristics and provision of a sufficient supply of auxiliary feedwater. The system design and pump selection shall also address upper limits on flow that might be imposed to mitigate the effects of mass and energy release inside containment and prevent reactor system cooldown at an excessive rate.

To comply with flow requirements and limits and at the same time avoid system complications that might result in decreased reliability, pump selection requires optimization within the framework of a specific system concept. The major performance requirements necessary to achieve a viable system design and pump selection are outlined in this section.

5.1.1 Flow requirements

The following guidance is provided to ensure the adequacy of AFS flow. As such, the guidance is conservative. Excessive conservatism, which would be detrimental to efficient system performance (e.g., flow control valve sizing), should be avoided, taking into account the realistic operating range.

5.1.1.1 Minimum flow

Minimum recirculation flow refers to the minimum actual AFS pump flows during system operation. At times the required auxiliary feedwater flow to be delivered is zero but the AFS pumps are started and put in a minimum recirculation flow configuration. Hence, minimum recirculation flow is the flow required to ensure AFS pump operability when the delivered flow is zero.

- (A) Minimum recirculation flow shall be established on the basis of the flow required to preclude pump and motor damage if the pump is started against a closed system [i.e., one in which flow cannot be delivered to the steam generator(s)].

The AFS pump(s) might be started either automatically or manually against a closed system. The AFS pump minimum recirculation flow configuration shall accommodate the worst-case system configuration, pump start or stop due to operator error, and performance differences between supposedly identical pumps and shall also allow the AFS pump(s) to operate without damage or protective trip. Minimum recirculation flow configurations shall be evaluated with the AFS for applicable postulated single failures [4]. Manual pump shutdown by the operators shall not be relied upon to protect the AFS pump(s) unless supported by available indication, procedures, and engineering evaluation (see Sec. 5.1.3.1).

Pump minimum flow requirements shall be established to ensure adequate pump cooling due to fluid temperature rise and to protect against pump cavitation due to hydraulic instability.⁵⁾

⁵⁾ NRC Bulletin 88-04 provides guidance relative to minimum flow requirements [7].