American Nuclear Society

REAFFIRMED

July 23, 2002
ANSI/ANS-51.10-1991
(R2002)

auxiliary feedwater system for pressurized water reactors

an American National Standard

This standard has been reviewed and reaffirmed with the recognition that it may reference other standards and documents that may have been superseded or withdrawn. The requirements of this document will be met by using the version of the standards and documents referenced herein. It is the responsibility of the user to review each of the references and to determine whether the use of the original references or more recent versions is appropriate for the facility. Variations from the standards and documents referenced in this standard should be evaluated and documented.
This standard does not necessarily reflect recent industry initiatives for risk informed decision-making or a graded approach to quality assurance. Users should consider the use of these industry initiatives in the application of this standard.

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American National Standard Auxiliary Feedwater System for Pressurized Water Reactors

Secretariat
American Nuclear Society

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American National Standard

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Addendum to Foreword

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Auxiliary Feedwater System for Pressurized Water Reactors

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Standards Committee (NFSC) with the recognition that it may reference other standards and documents that may have been superseded or withdrawn.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

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Foreword

(This Foreword is not a part of American National Standard for Auxiliary Feedwater System for Pressurized Water Reactors, ANSI/ANS-51.10-1991.)

This standard is applicable to pressurized light water reactor nuclear power plants.

The standard 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 revisions incorporated in this issue of the standard is station blackout (i.e., the loss of all alternating current power sources). A requirement is included that the system be capable of operating for a plant specific duration with the loss of all alternating current 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 50.63, "Loss of All Alternating Current Power." In the 1979 edition of this standard, a generic duration of two hours was identified and no rationale was included.

The membership of Working Group ANS-51.10 of the Standards Committee of the American Nuclear Society during the development of this document was:

J. F. Garibaldi, Chairman, *Ebasco Services, Inc.*
D. G. Keith, *Bechtel Power Corporation*
S. Ku, *GPU Nuclear Corporation*
K. J. Vehstedt, *New York Power Authority*
W. T. LeFave, *U.S. Nuclear Regulatory Commission*

In addition, the following individuals assisted in the preparation of this document:

M. A. Anzalone, *Ebasco Services, Inc.*
W. J. McTigue, *Ebasco Services, Inc.*
J. S. Pfabe, *Ebasco Services, Inc.*
D. Santory, *Ebasco Services, Inc.*

The membership of MC-1, LWR Design Criteria Standards Management Committee, at the time of its concurrence with the technical approval was:

W. H. D'Ardenne, Chairman, *General Electric Company*
E. J. Borella, *Ebasco Services, Inc.*
D. M. Crowe, *Georgia Power Company*
R. Fortier, *Stone & Webster Engineering Corporation*
J. C. Glynn, *U.S. Nuclear Regulatory Commission*
P. Hepner, *Combustion Engineering*
D. G. Keith, *Bechtel Power Corporation*
H. G. O'Brien, *Tennessee Valley Authority*
H. C. Shaffer III, *Yankee Atomic Electric Company*
R. C. Surman, *Westinghouse Electric Corporation*

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S. L. Stamm ..... Stone & Webster Engineering Corporation
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T. J. Sullivan ..... Institute of Nuclear Power Operations
C. D. Thomas ..... Yankee Atomic Electric Company
W. T. Ullrich ..... Philadelphia Electric Company
(for the American Nuclear Society)
G. P. Wagner ..... Commonwealth Edison Company
G. L. Wessman ..... Consultant
G. J. Wrobel ..... Rochester Gas & Electric Corporation

Contents

Auxiliary Feedwater System for Pressurized Water Reactors

1. Introduction and Scope

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

1.2. Purpose. This standard is written for new facilities. The 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 nuclear safety-related AFS. The standard also discusses relevant requirements associated with the AFS instrumentation and controls, source of power, water supply, and support systems interfaces.

2. Definitions

active failure. A malfunction, excluding passive failures, of a component that relies on mechanical movement to complete its intended nuclear safety function upon demand.

engineered safety feature (ESF). A safety class structure, system, or component that serves to control and limit the consequences of releases of energy and radioactivity in the event of Plant Condition 3, 4, or 5 events to the extent that the public health and safety might be impaired if these energy and radioactivity releases were not additionally restrained.

hot standby. The condition in which the reactor is subcritical and the reactor coolant system average temperature is above the required temperature to permit operation of the low pressure residual heat removal system, consistent with technical specification operational limits.

initiating occurrence. A single occurrence and its consequential effects that place the plant or some portion of the plant in an off-normal condition. An initiating occurrence is not the single failure defined in this section. An initiating occurrence can be an equipment failure, a human error, a natural hazard, or a man-made hazard.

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.

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

no loss of nuclear safety function. The capability of unit structures, systems, and components to accomplish nuclear safety functions required to accommodate normal operation or a specified event within applicable nuclear safety criteria.

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

orderly shutdown and cooldown. A shutdown and cooldown in which the fuel and reactor coolant pressure boundary conditions are within technical specification operational limits. Automatic actuation of an engineered safety feature may be required.

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 structures, systems, and components that are designed to perform at least one function during or after a safe shutdown earthquake (SSE).