American Nuclear Society

general safety design criteria for a liquid metal reactor nuclear power plant

an American National Standard

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ERRATA

American National Standard General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant, ANSI/ANS-54.1-1989

Page 9, Subsection 3.4.10, Structural and Equipment Cooling Systems, final paragraph, first sentence:

The word "not" should replace the word "now" in the first sentence of final paragraph of subsection 3.4.10; it should read:

> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system nuclear safety function can be accomplished, assuming a single failure.

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American National Standard General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant

Secretariat
American Nuclear Society

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

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American National Standard

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Comments on this standard are encouraged and should be sent to Society Headquarters.

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Foreword

(This Foreword is not a part of American National Standard for General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant, ANSI/ANS-54.1-1989.)

Pursuant to the provisions of Title 10, Part 50, Section 50.34 of the Code of Federal Regulations, an application for a nuclear power plant construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing and performance requirements for structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

General design criteria, which establish the minimum requirements for the principal design criteria, for nuclear power plants are identified in the Code of Federal Regulations, Title 10, Part 50, Appendix A (10CFR50A). While these criteria provide guidance for all types of nuclear power plants, they are specifically oriented toward water reactors. This is recognized in the Code of Federal Regulations which states: "These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units."

As a result of the increased design and development activities directed toward the establishment of commercial liquid metal reactor (LMR) plants, the need for more specific guidance for the design of these plants was recognized. Consequently, the American Nuclear Society Subcommittee ANS-24 (subsequently renamed Working Group ANS-54.1) was established in 1970 to develop and interpret these criteria for the LMR. The working group included representatives from the reactor and architect-engineer vendors, utilities, and the Atomic Energy Commission's regulatory and development divisions (later renamed the Nuclear Regulatory Commission and the Department of Energy). The efforts of this group resulted in General Safety Design Criteria for an LMFBR Nuclear Power Plant. Early in 1975, the group balloted in favor of issuing these criteria for trial use and comment, and ANSI N214, General Safety Design Criteria for an LMFBR Nuclear Power Plant was issued for trial use and comment in April 1975. The application of these criteria resulted in comments, which were addressed by the group. Subsequent comments by ANS-54 and by the Nuclear Power Plant Standards Committee (NUPPSCO) were addressed, but the proposed standard was withdrawn so that later information could be incorporated and the standard could be made applicable to large loop and pool type designs as well as small modular reactors. In developing the current standard, emphasis was placed on retaining the 10CFR50, Appendix A criteria without change wherever applicable to LMRs. Changes and/or additions were made to reflect the unique characteristics of LMRs. The Clinch River Breeder Reactor Plant licensing experience, which demonstrated one acceptable approach for licensing LMRs by meeting all the requirements for a Construction Permit, was factored into this standard. The standard also provides alternatives for controlling the risk of loss of core coolable geometry. In the process of providing these alternatives, the inherently safe core configurations and passive heat removal systems being developed under the advanced LMR program were considered. This resulted in the current standard.

The ANS-54.1 Working Group which prepared the final version of this standard consisted of the following membership:

L. E. Strawbridge, Chairman, *Westinghouse Electric Corporation*
Q. L. Baird, *Hanford Engineering Development Laboratory*
G. Berg, *Rockwell International Corporation*
C. S. Ehrman, *Burns and Roe, Inc.*
O. E. Gray, *Consolidated Management Office for the LMFBR*
J. R. Humphreys, *U.S. Department of Energy*
T. L. King, *U.S. Nuclear Regulatory Commission*
W. R. Rolf, *Commonwealth Edison Company*

At the time of approval of this standard, the American Nuclear Society Standards Subcommittee ANS-54 consisted of the following membership:

R. T. Lancet, Chairman, *Rockwell International Corporation*
R. F. Stearns, Secretary, *Bechtel Group, Inc.*
H. Alter, *U.S. Department of Energy*
C. Bijlani, *Burns & Roe, Inc.*
H. W. Buschman, *Argonne National Laboratory-West*
C. Cox, *Westinghouse-Hanford*
G. L. Gyorey, *GE Advanced Nuclear Technology*
T. L. King, *U.S. Nuclear Regulatory Commission*
H. R. Michael, *Stone & Webster Engineering Corporation*
M. Natta, *Institute de Protection et de Surete Nucleaire (France)*
J. M. Rich, *Sargent & Lundy*
W. R. Rolf, *Commonwealth Edison Company*
W. J. Rowan, *Consultant*
L. E. Strawbridge, *Westinghouse Electric Corporation*

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

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

W. M. Andrews	Southern Company Services, Inc.
F. Boorboor	United Engineers & Constructors*
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R. E. Miller	Duke Power Company
J. A. Nevshemal	Toledo Edison Company
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S. L. Stamm	Stone & Webster Engineering
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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
G. P. Wagner	Commonwealth Edison Company
G. L. Wessman	Consultant
G. J. Wrobel	Rochester Gas & Electric Corporation

*Affiliation at time of balloting

Contents

General Safety Design Criteria for a Liquid Metal Reactor Nuclear Power Plant

1. Scope and Introduction

1.1 Scope. This standard establishes General Safety Design Criteria which constitute the minimum safety requirements for liquid metal reactor (LMR) nuclear power plants and are intended to be used for the design of LMRs in lieu of the General Design Criteria for Nuclear Power Plants in the Code of Federal Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants" [1].[1] These criteria are intended to be applicable to LMR designs which use various arrangements of components, such as those characterized as loop designs, pool designs or combinations thereof. These reactors use sodium in the reactor coolant system and operate with a fast (high energy) neutron spectrum.

1.2 Introduction. This standard is intended to maintain a level of safety for LMRs at least equivalent to that established for LWRs by the General Design Criteria of 10CFR50, Appendix A [1]. The level of detail specified in this standard has been maintained comparable to that in 10CFR50, Appendix A [1].

These criteria also establish the design option to either prevent loss of core coolable geometry from specified beyond the design basis events so that design measures to mitigate such loss are not required, or to mitigate the consequences of such loss. To prevent loss of core coolable geometry, acceptably high reliability must be provided for the reactor shutdown systems and heat removal systems or it must be shown that failure of such systems does not result in loss of core coolable geometry. Preventive or mitigative features shall be provided to ensure acceptable risk for beyond the design basis events including anticipated transients without scram, station blackout and fuel failure propagation.

[1]Numbers in brackets refer to corresponding numbers in Section 4, References.

Identification of which structures, systems and components referred to in these criteria are nuclear safety-related shall be determined on a case-by-case basis for the specific reactor design being considered. The term "nuclear safety-related" is used throughout this standard to emphasize the protection of the public under postulated accident conditions. This standard does not address safety classification. It is recognized that such classification could include additional considerations related to protection of the public health and safety.

Environmental effects of nonradioactive toxic materials are not considered in these criteria since normal industry standards and practice are applicable.

2. Explanations and Definitions

2.1 Explanations. Definitions developed by the American Nuclear Society Glossary of Terms in Nuclear Science and Technology [2] have been used to the extent they are applicable to LMRs. Some definitions were taken directly from 10CFR50, Appendix A [1]. Because of the need to develop some definitions which are unique to LMRs, the following definitions are a composite set applicable specifically to this standard.

2.2 Definitions

active component. A mechanical or electrical component that has moving parts or an electrical component that is designed to perform its function by a change of configuration or properties.

active component failure. A malfunction, excluding passive failures, of an active component that would prevent completion of its intended function upon demand.

anticipated operational occurrences. Those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to a loss of all offsite power, an inadvertent control rod withdrawal, and tripping of the turbine generator set.