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DESIGN GUIDE
FOR
CATEGORY V REACTORS
TRANSIENT REACTORS

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

UNDER CONTRACT NO. EY-76-C-02-0016 WITH THE
UNITED STATES DEPARTMENT OF ENERGY

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DESIGN GUIDE FOR CATEGORY V REACTORS TRANSIENT REACTORS

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March 1979

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Prepared for the
DIVISION OF OPERATIONAL AND ENVIRONMENTAL SAFETY

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The Department of Energy (DOE) is the successor to the Energy Research and Development Administration (ERDA) and all references to ERDA herein shall be deemed to refer to DOE.

Printed in the United States of America
Available from

National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

Price: Printed Copy \$9.00; Microfiche \$3.00

April 1979

265 copies

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FOREWORD

The Department of Energy (DOE) owns and operates a wide variety of nuclear reactors and critical facilities as part of a comprehensive program which supports the development of advanced energy systems. As required by the ERDA Manual, Chapter 0540, DOE-owned reactors are sited, designed, constructed, modified, operated, maintained, and decommissioned with adequate consideration given to health and safety factors. In achieving these goals, the ERDA Manual further requires the use of generally uniform standards, guides, and codes that are comparable to those applied to similar reactors licensed by the Nuclear Regulatory Commission (NRC).

Because of significant differences between most DOE-owned reactors and NRC-licensed commercial power reactors, the design and operating requirements for the DOE reactors are not necessarily the same as those that apply to commercial reactors. In general, the DOE-owned reactors present less risk to health and safety than do commercial power reactors because of factors such as remote siting, lower power levels, lower fission product inventories, and reduced dependence on the operation of systems required to assure safe reactor shutdown and cooling and to assure acceptable consequences of postulated accidents. As a result, simplified and/or unique DOE design and performance requirements achieve the same defense-in-depth characteristics as the larger, more complicated commercial power reactors.

DOE-owned reactors vary considerably in their design, operating, and siting characteristics and in their programmatic goals. A method has been developed for systematically and uniformly recommending codes, standards, guides, and general design criteria that apply to each facility in the diverse spectrum of DOE-owned reactors. The method is based on grouping the DOE-owned reactors into categories that are defined by generic reactor types. Within a given category, the DOE-owned reactors have generally similar design and/or operating characteristics. Table A-1 defines six generic categories of DOE reactors. Table A-2 lists the DOE reactors that are considered to be Category V reactors.

For each reactor category, a Design Guide is being developed. This document is tailored to provide design and safety guidelines for systems typical of reactors within the particular category. The Design Guides are written to provide a DOE reactor facility contractor with recommendations to be considered in the design or modification of typical reactor facility systems. For each system, guidance is provided for determining if the system is required at a particular facility, and when the system is required, the Design Guide recommends major safety concepts and design philosophies to be implemented in the system in order to adequately protect health and safety.

In addition to the guidelines applicable to specific systems, guidelines are provided for reactor facility siting, general component and system design, conduct of operations, initial test programs, accident analysis, technical specifications and quality assurance. For each reactor system, and for each of the general topical areas described above, the Design Guide recommends industry

Table A-1

ERDA Reactor Categories

Category	General Reactor Characteristics	
	Type	Sub-Type
I	Critical Facilities	Solid Fuel Systems
		Liquid Fuel Systems
		Gaseous Fuel Systems
II	Water Cooled Reactors with a Closed Primary Coolant System	Tank-Type Reactors
		PWRs
		Water-Cooled Graphite Reactors
III	Pool-Type Reactors	--
IV	Liquid Metal Cooled Reactors	--
V	Transient Reactors	Fast Burst-Type Reactors
		Pulse-Type ⁽¹⁾
VI	Air Cooled Graphite Reactors	--

(1) This Category V reactor sub-type includes multi-mode (steady state and pulsing) tank-type, pool-type and graphite reactors which are also included in other Categories. The Design Guide for Category V reactors provides guidelines pertaining to the transient-related aspects of the facility. The steady-state aspects of the facility are included in the Design Guides for other category reactors which also pertain to reactors without transient capabilities.

Table A-2

DOE Category V Reactors

Category V Reactor Subgroup	DOE Reactor	Reactor Location	Remarks
Fast Burst Reactors	SPR II SPR III Super Kukla HPRR Godiva IV SKUA SAREF	Sandia Sandia NTS, Nevada ORNL LASL LASL INEL	- - - - - - -
Other Pulsing Reactors (1)	PBF ACRR TREAT	INEL Sandia INEL	Tank-Type Reactor, See Design Guide for Category II Reactors Pool-Type Reactor, See Design Guide for Category III Reactors Air-Cooled Graphite Reactor, See Design Guide for Category VI Reactors

- (1) The Design Guide for Category V Reactors provides guidelines pertaining to the transient-related aspects of these facilities. Other aspects of these facilities are covered by the Design Guide referenced in the remarks column.

and NRC standards, codes and guides, and the applicable portions of 10 CFR (including the General Design Criteria in 10 CFR 50, Appendix A) to be considered by the DOE reactor facility contractor.

Some existing nuclear industry standards and codes and USNRC Regulatory Guides, although written primarily for commercial power reactors, are also generally applicable to the design of certain Category V reactors. The Design Guide includes guidelines which provide interpretations of the standards, codes, and guides in order to facilitate their application of Category V reactors. When criteria from existing standards, codes, and guides do not provide adequate guidance, the guidelines in the Design Guide include interim recommendations for the designer pending the development or approval of appropriate new standards, codes, and guides.

Aided by the introductory recommendations in the Design Guide, it remains for the designer to consult the appropriate referenced standards, codes, and guides in order to be fully appraised of all system design requirements. All of the applicable standards, codes, and guides that are recommended for consideration by the designer are available from the appropriate issuing organization.

A program will be developed to periodically update and revise the Design Guide, as necessary, to reflect future standards, revisions, and developments and to incorporate changes recommended by the Design Guide users. To facilitate feedback from users of this guide, Chapter 18 includes preaddressed comment sheets that may be used to transmit comments or suggestions for Design Guide changes. In this manner, the Design Guide will remain a timely standard reference for the designer and reviewer alike.

CHAPTER 1

INTRODUCTION

The Department of Energy (DOE), as stated in the ERDA Manual, requires that all DOE-owned reactors be sited, designed, constructed, modified, operated, maintained, and decommissioned in a manner that gives adequate consideration to health and safety factors. Specific guidance pertinent to the safety of DOE-owned reactors is found in Chapter 0540 of the ERDA Manual.

1.1 PURPOSE

The purpose of this Design Guide is to provide additional guidance to aid the DOE facility contractor in meeting the requirement that the siting, design, construction, modification, operation, maintenance, and decommissioning of DOE-owned reactors be in accordance with generally uniform standards, guides, and codes which are comparable to those applied to similar reactors licensed by the Nuclear Regulatory Commission (NRC).

1.2 SCOPE

This Design Guide deals principally with the design and functional requirements of Category V reactor structures, components, and systems.

1.2.1 General Category V Reactor Characteristics

The guide is specifically applicable to reactors designed for performing controlled transients initiated by rapid reactivity insertion. The reactor is designed to limit the reactivity transient, generally by means of an inherent prompt negative reactivity coefficient.

1.2.2 Category V Reactor Subgroups

Category V reactors are divided into two subgroups: fast burst reactors and all other pulsing reactors.

Fast burst reactors characteristically use highly enriched uranium alloy fuel. These reactors are primarily designed for producing an intense radiation burst. Following a burst, the reactor core generally reverts to a subcritical geometry due to the mechanical disassembly of the core by the mechanical shock of the burst or by some alternate means. Fast burst reactors are also capable of limited steady-state, low-power operation. Due to their limited power histories, these reactors do not require active cooling systems during operation or following reactor shutdown. The small fission product inventory of fast burst reactors provides an inherent safety feature by limiting the potential radiological consequences of postulated accidents. This Design Guide is wholly applicable to fast burst reactors.

Other pulsing reactors include tank-type, pool-type, and air-cooled graphite reactors with the capability for producing controlled reactivity transients in addition to their capability for sustained operation at substantial power levels. This Design Guide only addresses the transient-related aspects of these facilities. Other Design Guides should be consulted for the basic design of safety features relevant to these facilities. Tank-type reactors are covered by the Category II Design Guide, pool-type reactors by the Category III Design Guide, and air-cooled graphite reactors by the Category VI Design Guide.

1.3 THE DESIGN GUIDE

1.3.1 Design Guide General Safety Philosophy

The design of DOE-owned reactors is based on fundamental safety philosophies comparable to those applied to NRC licensed commercial power reactors. Both require that the reactor be capable of being safely shut down and adequately cooled following postulated accidents. In addition, the reactor facility is designed to provide a defense-in-depth against the uncontrolled release of radioactivity to the environment.

Because of their potentially large inventories of radioactive material available for release, commercial power reactors provide multiple physical barriers to restrict the accidental release of radioactivity to the environment. These barriers generally include the fuel cladding, the reactor coolant system pressure boundary, and a containment system. Because of significant differences between Category V DOE-owned reactors and commercial power reactors, it is possible to simplify the design of the DOE reactor facility while still adequately protecting health and safety. In general, Category V reactors present much less risk to health and safety than do commercial power reactors because of such factors as remote siting, much lower potential source terms, and reduced dependence on the operation of systems required to assure safe reactor shutdown and/or to prevent an uncontrolled release of radioactivity to the environment. As a result, Category V DOE reactor facilities require fewer physical barriers to the release of radioactivity to the environment than commercial power reactors in order to remain within the off-site and on-site radiological consequence guidelines (see Table 1.3-1). For example, Category V DOE reactor facilities generally utilize a reactor building that does not control the release of radioactivity to the environment as a containment would for commercial power reactors. Defense-in-depth is provided at these DOE facilities by inherent mechanisms for terminating reactivity bursts and shutting down the reactor, by their low fission product inventory and consequent low shutdown heat generation rate (which eliminates the need for active shutdown core cooling systems). These inherent safety features provide the required reasonable assurance that the DOE-owned Category V reactor facility can be operated without undue risk to health and safety.

Table 1.3-1 provides a summary of the general safety philosophy for operation in terms of frequency of occurrence and radiological consequence guidelines corresponding to each facility operating condition (normal, upset, emergency and

Table 1.3-1

Summary of the General Safety Philosophy for
Facility Operation in Terms of Frequency of
Occurrence and Radiological Consequence Guidelines

Operating Condition	Description	Typical Design Basis Events	Radiological Consequence Guidelines	
			Off-Site	On-Site
Normal	Events that are planned to occur regularly in the course of plant operation	Startup, shutdown, normal reactivity pulses	ALARA* ≤ERDAM 0524	ALARA* ≤ERDAM 0524
Upset	Incidents or occurrences of moderate frequency which may occur several times during the life of a facility	Loss of electric power, inadvertent reactor shutdown	ALARA* ≤ERDAM 0524	ALARA* ≤ERDAM 0524
Emergency	Infrequent incidents or occurrences which may occur during the lifetime of a facility	Reactivity burst larger than planned	Small fraction of 10CFR100	Small fraction of on-site faulted dose guidelines
Faulted	Events which are limiting faults and are not expected to occur during the life of a facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material and because they represent upper bounds on failures or accidents with a probability of occurrence sufficiently high to require consideration in design. Generally a postulated accident is not considered in design if the probability of its occurrence resulting in unacceptable consequences is less than 10^{-7} per year	Design basis reactivity accident	≤10CFR100	≤25 rem whole body, ≤300 rem thyroid, ≤150 rem bone, ≤75 rem lung, and ≤75 rem to other organs

*As low as reasonably achievable

faulted). The facility operating conditions are discussed in detail in Section 3.9 and in Chapter 15. The radiological consequence guidelines in Table 1.3-1 are to be interpreted as minimum guidelines to protect health and safety. Each DOE contractor has the additional responsibility for maintaining radiation exposures As Low As Reasonably Achievable (ALARA) - (see Chapter 12).

A. Safety Class and Safety Function

Category V reactor structures, components, and systems that are required for the protection of health and safety are those required for assuring safe shutdown and/or to prevent the uncontrolled release of radioactivity to the environment. These structures, components, and systems have certain essential performance characteristics called safety functions that should generally be implemented in the system design in order to provide the systems capabilities necessary for adequate protection of health and safety. Because of the varied designs of Category V reactors, it is not within the scope of this Design Guide to specify all of the safety functions that apply to each system at a specific facility. The safety functions that are listed in this Design Guide in Chapters 4 to 12 are those that are customarily associated with each particular system. Each DOE contractor may determine the applicability of the safety functions, listed in this Design Guide, to his own reactor facility. Additionally, the DOE contractor should determine if other safety functions, not identified by this Design Guide, may be applicable to his reactor facility.

The implementation and maintenance of the safety functions is achieved through the design of components, systems and structures to quality standards that are commensurate with the importance of their safety function(s). A graded set of quality standards establish the basis for defining the design requirements that apply to each of three nuclear safety classes (Safety Classes 1, 2, and 3) and to a non-safety class (non-nuclear).

The design guidelines presented in this Design Guide for determination of the applicable safety class and other design features are those that are generally considered necessary for the implementation and maintenance of the safety functions. It is not within the scope of this Design Guide to specify the safety class and all guidelines that are applicable to each system at a specific facility. Each DOE contractor may determine the suitability of the recommended minimum safety class to his own reactor facility. Additionally, the DOE contractor should determine if other design guidelines, not identified by this Design Guide, and/or a higher final safety class may be applicable to his reactor facility.

B. Designation of Safety Class Structures, Components and Systems

This Design Guide provides recommendations regarding the minimum safety class to which structures, components, and systems should be designed. Additionally, guidelines are provided to assist the DOE contractor in determining the final safety class to which structures, components, and systems at a specific reactor facility should be designed.

1. Minimum Safety Class

The minimum safety classes recommended by this Design Guide are based on USNRC Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants." If the maximum conservatively calculated off-site doses due to a postulated failure of the system may exceed 0.5 rem to the whole body, the system, or appropriate portions of it, should be designed to Safety Class 1, 2, or 3. If the maximum conservatively calculated off-site dose due to system failure will not exceed 0.5 rem, it may be designed to non-nuclear standards. Additional guidelines for determining minimum safety class are found in Section 3.2 and in Chapters 4-12.

2. Final Safety Class

Based on the accident analysis performed during the Safety Analysis Report (PSAR or FSAR) stage of facility review, the adequacy of the minimum safety class recommended by the Design Guide can be determined. Because of specific features of an individual Category V reactor facility, analysis of some design basis accidents may indicate that the on-site and/or off-site radiological consequences exceed the guidelines in Table 1.3-1. Therefore, some system design changes may be necessary to ensure that a final safety class, equivalent to the recommended minimum safety class, will result in consequences within the guidelines in Table 1.3-1. Alternatively, it may be necessary to increase the quality and reliability of the system by designing it to a final safety class that is higher than the minimum safety class recommended by this Design Guide.

It is the responsibility of the DOE contractors to determine the final safety class of structures, systems, and components at specific reactor facilities in cases where a safety class different than the minimum safety class recommended by this Design Guide is required for either safety or programmatic reasons. Additional guidelines for determining final safety class are found in Chapter 15.

1.3.2 Use of the Design Guide

This Design Guide has been written to provide standardized guidelines and recommendations pertinent to the design of new reactor facilities and to the modification of significant portions of existing facilities. For each system, the Design Guide provides guidance for determining if the system is required at a particular facility and, if the system is required, the Design Guide recommends major safety concepts and design philosophies and the codes, standards, and guides to be considered in system design and/or modification.

Some existing industry standards and codes and USNRC Regulatory Guides, although written primarily for commercial power reactors, are also generally applicable to the design of Category V reactors. Where necessary, design guidelines provide interpretations of the standards, codes, and guides in order to facilitate their application to Category V reactors. When existing standards, codes, and guides do not provide adequate guidance, the guidelines presented in

this Design Guide serve as interim recommendations pending the development or approval of appropriate new standards, codes, and guides.

Aided by the introductory guidance provided by this Design Guide, it remains for the designer to consult the appropriate referenced standards, codes, and guides in order to be fully appraised of all system design requirements. All of the applicable standards, codes, and guides which should be considered by the designer are readily available from the appropriate issuing organization.

As necessary, to reflect programmatic requirements, DOE Headquarters or a DOE Field Office may identify supplementary standards or design guidelines to be used by a DOE contractor for reactor design or modification. DOE Headquarters or the appropriate DOE Field Office may also approve the design or modification of a reactor facility to standards, codes, and guides other than those recommended by this Design Guide if the DOE contractor can demonstrate that the recommendations of this Design Guide are overly restrictive for his particular application and/or that the alternate design guidelines adequately protect health and safety.

1.3.3 Design Guide Organization

Figure 1.3-1 illustrates the organization of this Design Guide and is the starting point for determining the portions of this guide that should be consulted when designing a new reactor facility or modifying an existing reactor facility. As illustrated in Figure 1.4, the design of any system should consider the guidance provided in Chapter 3 and in the appropriate chapter or section which presents the detailed system design guidelines.

Table 1.3-2 lists the systems that are typically associated with Category V fast burst reactors and also references the Design Guide chapter or section that provides the detailed system design information. Also listed in Table 1.3-2 are the systems of other pulsing reactors that may require special design provisions due to the transient-related aspects of plant operation. The basic design and safety guidelines for other pulsing reactors are covered in other Design Guides, as described in Section 1.2. The Category V Design Guide provides additional guidelines applicable to other pulsing reactor systems listed in Table 1.3-2.

The recommendations for each system are presented in an individual chapter or section of this Design Guide (Chapters 4-12) which includes the following information:

- A. Description of the system safety function
- B. Definition of the system
- C. A diagram that provides a guide to the use of the chapter or section.
- D. Description of the rationale for determining the minimum system safety class.

Figure 1.3-1. Design guide organization.

Table 1.3-2

Typical Systems for Category V Reactors

System	Applicable Design Guide Section	
	Fast Burst Reactors	Other Pulsing Reactors ⁽¹⁾
Reactor Core Assembly	4.1	4.1
Reactivity Control System	4.2	4.2
Reactor Building ⁽²⁾	6.2	--
Protection System ⁽³⁾	7.1	--
Non-Safety I&C System	7.2	--
Non-Class IE Electric System	8.2	--
Fuel Handling & Storage System	9.2	--
Fire Protection System	9.4	--
Experimental Facilities	10	10
Radioactive Waste Systems	11	--
Radiation Protection System	12	12

Notes: (1) Other Design Guides should be consulted for the basic design and safety guidelines for these facilities (e.g. Category II, III and VI Design Guides for tank-type, pool-type and air-cooled graphite reactors, respectively).

(2) Generally, the reactor building need not be designed as a containment.

(3) Generally for reactor operating interlocks and reactor trip function only.

E. General design guidelines

F. Listings of recommended General Design Criteria (10 CFR 50, Appendix A, and other 10 CFR requirements), industry codes and standards, RDT Standards, and USNRC Regulatory Guides that should be considered in the design of the system. This section includes industry standards that are approved and draft standards that have been issued for trial use and comment.

G. Listings of supplementary information that may provide useful background information to the designer. This section is a basic bibliography of pertinent references, including USNRC Standard Review Plans and Branch Technical Positions, IAEA Safety Guides, and other general references. Also included in this section are industry codes and standards that are proposed for future development or are distributed on only a limited basis as preliminary drafts.

H. Listings of areas requiring additional standards development. This section lists areas of design, operation, administration and other topical areas in which additional standards development (in addition to the existing and planned standards previously listed) may provide a more consistent approach to reactor facility design and modification.

In addition to the information applicable to specific systems or structures, this Design Guide also provides general guidelines and references for Siting (Chapter 2), General Component and System Design (Chapter 3), Conduct of Operations (Chapter 13), Initial Test Program (Chapter 14), Accident Analysis (Chapter 15), Technical Specifications (Chapter 16), and Quality Assurance (Chapter 17).

1.3.4 Maintenance of the Design Guide

The continuing development and revision of standards, codes, and guides that are applicable to Category V reactor facilities will necessitate the periodic updating of the Design Guide in order to ensure that the guidance being provided reflects the most current reactor safety developments. Additional revisions will be required to reflect comments and requests for additional information that are likely to arise during use of the Design Guide by DOE and the DOE facility contractors. A program will be established for the periodic revision of the Design Guide and for provision of additional information, as necessary, during the period between revisions.

1.4 DEFINITIONS

Definitions of many terms used in the Design Guide are given below:

A. ALARA

Pertaining to maintenance of personnel exposure during routine operation of a reactor facility to radiation levels "as low as reasonably achievable" (10 CFR 50, Appendix I and Regulatory Guide 8.8). This term is comparable to ALAP (As Low As Practicable) discussed in ERDAM 0524.

B. Class IE

The safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, reactor core cooling, or otherwise are essential in preventing a significant release of radioactive material to the environment (IEEE 308-1974, Revised).

C. Controlled Area

Any area, access to which is controlled for the purpose of protection of individuals from exposure to radiation and radioactive materials (ERDAM 0524).

D. Design Basis Accidents

Those limiting accidents of low probability that are postulated and analyzed to establish design requirements of all safety class equipment (ANSI N213).

E. Health and Safety

Referring to the well being of the general public, DOE, and DOE contractor personnel and their protection from the potential hazards that may arise from the operation of DOE-owned reactors (ERDAM Chapter 0540-04, Revised).

F. Off-Site Dose

Potential radiological exposure to persons at or beyond the exclusion area boundary defined in 10 CFR 100.3.

G. On-Site Dose

Potential radiological exposure to persons within the exclusion area boundary.

H. Operating Conditions

The grouping of events that may occur at a reactor facility (including normal, upset, emergency, and faulted events) based on frequency of occurrence and severity of event, for the purpose of determining the consequence guidelines are applicable to the particular event.

I. Safety Class

The grouping of reactor facility components, systems and structures based on the importance of their safety functions, for the purpose of determining the applicable design and quality standards (corresponds to Quality Groups defined in USNRC Regulatory Guide 1.26).

J. Safety Function

Performance characteristics required to assure safe shutdown and cooling of a reactor and/or to prevent an uncontrolled release of radioactivity to the environment.

K. Safety Limit

A limit on an important process variable which is found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency (10 CFR 50.36c, Revised).

L. Seismic Category I

The safety classification of the structures, systems, and components that should be designed to remain functional if an earthquake occurs (USNRC Regulatory Guide 1.29, Revised).

M. Severe Natural Phenomena

Those conditions which are derived from the history and properties of the site and surroundings, which are the most severe, and which can be reasonably expected to have the potential for causing consequences in excess of those allowed under applicable guidelines.

N. Uncontrolled Area

Any area, access to which is not controlled for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential purposes (inferred from ERDAM 0524).

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CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Guidelines for Reactor Siting

A. The size of the facility site and the location of the facility within the site boundary should provide reasonable assurance that the off-site radiological consequence guidelines in Chapter 15 will not be exceeded following any postulated accident.

B. Highways, railways, and waterways which traverse the site should be sufficiently distant from facility structures that routine use of these routes is not likely to interfere with normal facility operation. Hazardous material shipments over these routes should be evaluated for potential impact on the reactor facility. Potential externally generated missile hazards are discussed in Section 3.5.

C. The "exclusion area boundary" to be considered in the radiological evaluation of postulated accidents at a facility should be within or coincide with the boundary of the site at which the facility is located. The programmatic importance of neighboring facilities and on-site radiological safety considerations additionally require the analysis of potential on-site doses from postulated accidents. Maximum dose guidelines for on-site personnel and persons at the exclusion area boundary are defined in Chapter 15.

D. The 10 CFR 100 guidance concerning the "Low Population Zone" surrounding a reactor facility and the "Population Center Distance" should be considered in siting a reactor facility. Maximum accident dose guidelines for persons at the outer boundary of the "Low Population Zone" are defined by Chapter 15.

E. In the event that the maximum conservatively calculated doses resulting from postulated accidents do not exceed 0.5 rem at the "Exclusion Area Boundary," determination of the "Low Population Zone" and the "Population Center Distance" and resulting potential doses in these zones may not be necessary.

2.1.2 Applicable Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the reactor site:

A. 10 CFR 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection"

B. 10 CFR 100, "Reactor Site Criteria"

2.1.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the evaluation of the geographic and demographic aspects of a reactor site:

A. Industry Standards and Codes

1. ANSI/ANS 15.7-1977 (379), "Research Reactor Site Evaluation"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites"
2. USNRC Regulatory Guide 4.2, "Preparation of Environmental Routes for Nuclear Power Stations"
3. USNRC Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations"
4. USNRC Regulatory Guide 4.11, "Terrestrial Environmental Studies for Nuclear Power Stations"

2.1.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 2.1.1, "Site Location and Description," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 2.1.2, "Exclusion Area Authority and Control," *ibid.*
3. USNRC Standard Review Plan 2.1.3, "Population Distribution," *ibid.*
4. USNRC Standard Review Plan 2.2.1 and 2.2.2, "Location and Routes, Description," *ibid.*
5. USNRC Standard Review Plan 2.2.3, "Evaluation of Potential Accidents," *ibid.*

6. Weinberg, A.M., "Demographic Policy and Power Plant Siting," CONF-711058-1, Oak Ridge National Laboratory, October 13, 1971

7. WASH 1294, "General Environmental Siting Guides for Nuclear Power Plants: Topics and Bases," USAEC, December 1973

8. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962

9. IAEA Safety Guide SG-S4, Draft, "Site Selection and Evaluation of Nuclear Power Plants with Respect to Population Distribution," International Atomic Energy Agency, Vienna

10. IAEA Safety Guide SG-55, Draft, "Man-Induced Events Related to Nuclear Power Plant Siting," ibid.

B. Pending Industry Standards and Codes

1. ANSI N634 (ANS 2.6), Proposed, "Guidelines for Determining Present and Future Population Distribution at Power Reactor Sites"

2.1.5 Topics for Future Standards Development

None

2.2 METEOROLOGY

2.2.1 Guidelines for Reactor Siting

A. The reactor site should be evaluated to establish the safety-related meteorological site parameters. These include the following:

1. Operating Basis Wind
2. Design Basis Tornado
3. Local Probable Maximum Precipitation
4. Maximum Snow Load
5. Dilution of Gaseous Effluents
6. Extreme Temperature Ranges

B. Unless adequate meteorological data exists, an on-site meteorological program should be established using suitably accurate and reliable meteorological sensors and a suitable system for data recovery. A single meteorological program, covering all reactors at a larger site, would adequately provide this monitoring capability.

C. In the short-term diffusion estimates used for calculating the radiological consequences of postulated accidents, consideration should be given to the guidance in TID-24190 regarding the acceptability of models and conservatism appropriate to the calculations.

D. In the long-term diffusion estimates used for calculating the radiological consequence of normal releases, consideration should also be given to the guidance in TID-24190 regarding the acceptability of models and conservatism.

2.2.2. Applicable Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the reactor site:

A. 10 CFR 51, "Licensing and Regulatory Policy and Procedure for Environmental Protection"

B. 10 CFR 100, "Reactor Site Criteria"

2.2.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the evaluation of the meteorological aspects of a reactor site:

A. Industry Standards and Codes

1. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

2. ANSI/ANS 15.7-1977 (N379), "Research Reactor Site Evaluation"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water-Cooled Nuclear Power Plants"

2. USNRC Regulatory Guide 1.23, "On-Site Meteorological Programs"

3. USNRC Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"

4. USNRC Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from LWRs"

5. USNRC Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations"

2.2.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

The Standard Review Plans listed below provide comprehensive meteorology-related bibliographies.

1. USNRC Standard Review Plan 2.3.1, "Regional Climatology," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 2.3.2, "Local Meteorology," *ibid.*
3. USNRC Standard Review Plan 2.3.3, "On-Site Meteorological Measurement Programs," *ibid.*
4. USNRC Standard Review Plan 2.3.4, "Short-Term (Accident) Diffusion Estimates," *ibid.*
5. USNRC Standard Review Plan 2.3.5, "Long-Term (Routine) Diffusion Estimates," *ibid.*
6. WASH 1294, "General Environmental Siting Guides for Nuclear Power Plants: Topics and Bases," USAEC, December 1973
7. U. S. Department of Commerce, "Climatic Atlas of the United States," Environmental Data Service, NOAA, June 1968
8. U. S. Department of Commerce, "Local Climatological Data and Comparative Data," Environmental Data Service, NOAA, Published Annually for all First-Order NWS Stations
9. U. S. Department of Commerce, "State Climatological Summary," Environmental Data Service, NOAA, Published Annually by State
10. Holzworth, G. C., "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Office of Air Programs, USEPA, January 1972
11. Slade, D.H., Editor, "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC, 1968
12. Yanskey, G.R., Markee, E. H. and Richter, A. P., "Climatology of the National Reactor Testing Station," IDO-12048, Idaho Operations Office, USAEC, 1966

13. IAEA Safety Guide SG-S3a, Draft, "Meteorology-Climatology, Diffusion, and Transport," International Atomic Energy Agency, Vienna

14. IAEA Safety Guide SG-S3b, Draft, "Meteorology-Extreme Meteorological Conditions in Nuclear Power Plant Siting," ibid.

15. IAEA Safety Guide SG-S11b, Draft, "Design Basis Hurricane - Wind and Pressure Field," ibid.

B. Pending Industry Standards and Codes

1. ANSI N178 (ANS 2.3), Proposed, "Guidelines for Estimating Tornado, Hurricane and Other Extreme Wind Parameters at Power Reactor Sites"

2. ANSI N179 (ANS 2.5), Proposed, "Guidelines for Determining Meteorological Information Appropriate to a Power Reactor Site"

3. ANS 2.15, Draft, "Guidelines for Analysis of Atmospheric Transport of Routine Release of Nuclear Reactor Effluents"

4. ANS 2.16, Draft, "Guidelines for Analysis of Atmospheric Transport of Accidental Release of Nuclear Reactor Effluents"

2.2.5 Topics for Future Standards Development

None

2.3 HYDROLOGY

2.3.1 Guidelines for Reactor Siting

A. The potential for flooding from each of the following sources and events should be considered in evaluating the suitability of a reactor site: (For definitions, see USNRC Regulatory Guide 1.59).

1. Stream flooding.
2. Surges.
3. Seiches.
4. Tsunamis.
5. Seismically-induced dam failures (or branches) and maximum water level at the site.
6. Ice loadings from water bodies.

B. Groundwater-induced hydrostatic loadings should be considered in the design of subsurface portions of safety class structures.

2.3.2 Applicable Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the reactor site:

A. 10 CFR 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection"

B. 10 CFR 100, "Reactor Site Criteria"

2.3.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the evaluation of the hydrological aspects of a reactor site:

A. Industry Standards and Codes

1. ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites"

2. ANSI/ANS 15.7-1977 (N379), "Research Reactor Site Evaluation"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"

2. USNRC Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants"

3. USNRC Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I"

4. USNRC Regulatory Guide 1.135, "Normal Water Level and Discharge at Nuclear Power Plants"

5. USNRC Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations"

2.3.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

The Standard Review Plans listed below provide comprehensive bibliographies of hydrology-related references.

1. USNRC Standard Review Plan 2.4.1, "Hydrologic Description," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 2.4.2, "Floods," *ibid.*
3. USNRC Standard Review Plan 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers," *ibid.*
4. USNRC Standard Review Plan 2.4.4, "Probable Dam Failures (Seismically Induced)," *ibid.*
5. USNRC Standard Review Plan 2.4.5, "Probable Maximum Surge and Seiche Flooding," *ibid.*
6. USNRC Standard Review Plan 2.4.6, "Probable Maximum Tsunami Flooding," *ibid.*
7. USNRC Standard Review Plan 2.4.10, "Flood Protection Requirements," *ibid.*
8. USNRC Standard Review Plan 2.4.13, "Groundwater," *ibid.*
9. USNRC Standard Review Plan 2.4.14, "Technical Specifications and Emergency Operation Requirements," *ibid.*
10. WASH 1294, "General Environmental Siting Guides for Nuclear Power Plants: Topics and Bases," USAEC, December 1973
11. Reports of the Corps of Engineers, U. S. Army. (These reports address many specific hydrology-related topics.)
12. Hydrometeorological Reports and Technical Papers of the U. S. Weather Bureau (now U. S. Weather Service, NOAA), Hydrometeorological Branch. (These reports address many specific hydrology-related topics.)
13. IAEA Safety Guide SG-S10A, Draft, "Design Basis Floods for Nuclear Power Plants on River Sites," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

1. ANSI N515 (ANS 2.4), Proposed, "Guidelines for Determining Tsunami Criteria for Power Reactor Sites"

2.3.5 Topics for Future Standards Development

None

2.4 GEOLOGY AND SEISMOLOGY

2.4.1 Guidelines for Reactor Siting

The following geologic and seismic features should be considered in evaluating the suitability of a reactor site:

A. Geologic features. Mass-wasting, differential subsidence, faulting, soil and foundation instability, chemical weathering, cavernous or karst terrains, and volcanism.

B. Seismic features. Ground failure under dynamic loading, liquefaction, vibratory ground motion, site amplification, tsunamis, and residual stresses.

C. Man-made conditions. Subsidence or collapse caused by withdrawal of fluids or mineral extraction, and fault movement caused by fluid injection or withdrawal.

2.4.2 Applicable Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the reactor site:

A. 10 CFR 51, "Licensing and Regulatory Policy and Procedures for Environmental Protection"

B. 10 CFR 100, "Reactor Site Criteria"

2.4.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the evaluation of the geologic and seismic aspects of a reactor site:

A. Industry Standards and Codes

1. ANSI/ANS 15.7-1977 (N379), "Research Reactor Site Evaluation"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants"

2. USNRC Regulatory Guide 1.138, "Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants"

3. USNRC Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations"

2.4.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

The Standard Review Plans listed below provide comprehensive bibliographies of geology and seismology-related references.

1. USNRC Standard Review Plan 2.5.1, "Basic Geologic and Seismic Information," NUREG 75,087, September 1975
2. USNRC Standard Review Plan 2.5.2, "Vibratory Ground Motion," *ibid.*
3. USNRC Standard Review Plan 2.5.3, "Surface Faulting," *ibid.*
4. USNRC Standard Review Plan 2.5.4, "Stability of Subsurface Materials and Foundations," *ibid.*
5. USNRC Standard Review Plan 2.5.5, "Stability of Slopes," *ibid.*
6. Cohee, G. V., "Tectonic Map of the United States," U. S. Geological Survey and American Association of Petroleum Geologists, 1962
7. Historical Earthquake Data File, National Geophysical and Solar Terrestrial Data Center, NOAA
8. "Earthquake History of the United States," Publication 41-1, U. S. Department of Commerce, NOAA, 1973
9. Lomenick, T. F., "Earthquakes and Nuclear Power Plant Design," ORNL-NSIC-28, Oak Ridge National Laboratory, July 1970
10. IAEA Safety Guide SG-S1, Draft, "Earthquakes and Associated Topics for Nuclear Power Plants," International Atomic Energy Agency, Vienna
11. IAEA Safety Guide SG-S2, Draft, "Seismic Analysis and Testing of Nuclear Power Plants," *ibid.*

B. Pending Industry Standards and Codes

1. ANSI N18.4 (ANS 2.1), Proposed, "Guidelines for Determining the Vibratory Ground Motion for the Design Earthquake"
2. ANSI N174 (ANS 2.11), Proposed, "Guidelines for Evaluating Site-Related Geotechnical Parameters at Power Reactor Sites"

3. ANSI N180 (ANS 2.7), Proposed, "Guidelines for Assessing Capability for Surface Faulting at Power Reactor Sites"

2.4.5 Topics for Future Standards Development

None

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 INTRODUCTION TO DESIGN CLASSIFICATION

Category V reactor facility components, systems, and structures required for assuring safe shutdown of the reactor and/or for preventing an uncontrolled release of radioactivity with radiological consequences in excess of the guidelines in Chapter 15 should be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. In order to increase the level of confidence in these safety systems, and to allow credit to be taken for their capabilities in a conservative safety analysis, requirements are imposed in areas such as design, performance, operation and quality assurance. This chapter presents general guidance pertinent to the design of structures, components, and systems important to safety. The following chapters provide more detailed system design guidelines and additionally address the areas of facility operation and preoperational testing, accident analysis, technical specifications, and quality assurance.

Figure 3.1-1 is a guide to the use of Chapter 3.

3.1.1 Severe Natural Phenomena

As illustrated in Figure 3.1-1, the possible occurrence of severe natural phenomena should be considered in the design of safety systems when necessary to ensure the ability to safely shut down the reactor and/or to prevent uncontrolled release of radioactivity with consequences in excess of the guidelines in Chapter 15. Severe natural phenomena that may occur at a reactor site are discussed in Chapter 2. For a given site, it is possible to define the most severe natural phenomena that can reasonably be expected on the basis of the history and properties of the site and surroundings.

Because of unique design and operating characteristics of most Category V reactors, it may not be necessary to provide all safety systems with protection against all postulated severe natural phenomena. The acceptability of the reduced spectrum of severe natural phenomena against which Category V reactor safety systems need to be protected should be based on a demonstration of a correspondingly reduced period during which the reactor is "at risk" (e.g., capable of producing doses in excess of the guidelines in Chapter 15 or of being prevented from achieving a safe shutdown condition). The limiting severe natural phenomena (faulted events) to be considered in safety system design should be selected such that the joint probability of the occurrence of the phenomena during a time when the reactor is "at risk" is less than 10^{-6} to 10^{-7} .

Care should be taken in defining the time during which a reactor may be "at risk" since it may not be the same for all severe natural phenomena. For example, a fast burst reactor may only be "at risk" to some severe natural phenomena (e.g. tornado-generated missiles) during a reactivity transient. On the

Figure 3.1-1. Guide to the use of chapter 3.

* IN GENERAL, THE SYSTEM MAY FAIL AS A RESULT OF THE OCCURRENCE OF THE SEVERE PHENOMENA, HOWEVER, THE FAILURE MUST NOT ADVERSELY AFFECT THE ABILITY OF OTHER SYSTEMS TO PERFORM THEIR SAFETY FUNCTIONS WHEN REQUIRED.

Figure 3.1-1. Guide to the use of chapter 3 (continued).

other hand, the same facility may be "at risk" to flooding any time there is sufficient fissile material inventory at the facility to result in an inadvertent criticality when the reactor or the storage vault is flooded. When defining the spectrum of severe natural phenomena against which Category V reactor safety systems should be protected, the DOE contractor should establish the "at risk" period for the reactor facility subject to the approval of the cognizant DOE Field Office.

The periods during which a Category V reactor may be "at risk" to severe natural phenomena can be minimized by taking advantage of the fact that some severe natural phenomena, such as hurricanes and floods, can be forecast with sufficient accuracy and timeliness to allow some protective actions to be taken in advance of their most severe effects on a reactor facility. Note, however, that there are other severe phenomena, such as earthquakes, that cannot be accurately forecast and, therefore, may occur with little or no warning.

Unlike power reactors, which may be designed to continue operating during some severe natural phenomena in order to supply electric power, Category V reactors are operated intermittently on a flexible schedule that would permit shutting down the reactor when the facility is threatened by impending severe natural phenomena. At these facilities, some safety class equipment required during reactor operation (e.g., protection system) may not need to be protected from foreseeable severe natural phenomena if reactor shutdown is required in advance of the severe natural phenomena and the systems are then no longer required for safety (e.g., due to the scrammed condition of the reactor). Other safety class systems that are required for maintaining the reactor in a safe shutdown condition or for preventing an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15 should be protected against the effects of the severe natural phenomena.

At some facilities, technical specification limits on facility operation during the occurrence of severe natural phenomena may thus adequately protect health and safety and, thereby, provide an alternative to protecting some safety class equipment against the effects of the severe natural phenomena. Subsequent facility operation would, of course, be preceded by the satisfactory completion of inspections, repairs, and retesting of unprotected safety class components, systems, or structures that may have been damaged by the severe natural phenomena.

Category V reactor safety class components, systems, and structures should be designed to withstand severe natural phenomena that cannot be accurately forecast (e.g., earthquakes) to ensure that the system safety functions can be completed when necessary. Guidelines defining the spectrum of earthquakes to be considered in Category V reactor facility design are included in Section 3.2.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

Components, systems, and structures required to assure safe shutdown of a reactor and/or to prevent uncontrolled release of radioactivity with radiological consequences in excess of the guidelines in Chapter 15 are designated Seismic Category I, and are specifically designed to withstand the effects of certain earthquakes. Seismic Category I components, systems, and structures in commercial power reactors and in many DOE reactor facilities are designed to withstand the effects of the Safe Shutdown Earthquake (SSE). The SSE is based on an evaluation of the maximum earthquake potential of the reactor site and is that earthquake which produces the maximum vibratory ground motion for which safety class structures, components, and systems are designed to remain functional.

Because of the unique design and operating characteristics of most Category V reactors, their Seismic Category I components, systems and structures may provide adequate protection of health and safety when designed to a maximum earthquake, herein referred to as the Design Basis Earthquake (DBE), whose intensity is less than that of the SSE. Like the SSE, the DBE is based on an evaluation of the regional and local geology and seismology and of local subsurface materials. The acceptability of the reduced intensity of the DBE selected for a particular Category V reactor facility should be based on a demonstration of a correspondingly reduced period during which the reactor is "at risk" (e.g., capable of producing doses in excess of the guidelines in Chapter 15 or of being prevented from achieving a safe shutdown). The DBE is thus defined as an earthquake whose magnitude is selected such that the joint probability of its occurrence during a time when the reactor is "at risk" is less than 10^{-6} to 10^{-7} per year.

When defining a DBE for a Category V reactor facility, the following examples should be considered:

A. Fast-Burst Reactors

Consider a hypothetical facility that is "at risk" only during pulsing operations. On the basis of the expected reactor operating schedule, it is determined that this corresponds to an "at risk" period of approximately 10^{-5} per year. The seismic design of this facility may be based on a DBE with a probability of occurrence in the range of 10^{-1} to 10^{-2} per year. This would be a far more likely earthquake than the SSE and, thus, its magnitude should be much less than that of the SSE. The expected operating cycle used for defining the DBE should become part of the facility technical specifications and should not be exceeded without reevaluating and/or upgrading the seismic design of the facility as necessary.

B. Other Pulsing Reactors

These reactors are larger than fast burst reactors, may operate with a substantial power history and may have unique hazards associated with their

experimental facilities (e.g., mixed-oxide fuel test loops). As a result, these reactors may be "at risk" even when shut down (e.g., due to a significant decay heat level and fission product inventory or due to the nature of the experiments performed at the facility). These facilities may, therefore, be "at risk" for the majority of time during any given year. For conservatism, these facilities should be designed to withstand the effects of the SSE when the "at risk" period for the facility cannot be demonstrated to be $\leq 10^{-1}$ per year.

When defining a DBE for a Category V reactor facility, the DOE contractor should establish the "at risk" period for the reactor facility.

The use of a DBE as the basis for seismic design of a Category V reactor facility contrasts with the guidelines in 10 CFR 100, Appendix A, which postulates only an SSE or an Operating Basis Earthquake (OBE). The use of the DBE in Category V reactor design does, however, allow credit to be taken for inherent facility safety features while still satisfying the overall safety objective for an individual reactor facility; namely, to ensure that there is $\leq 10^{-6}$ per year chance of an accident with dose consequences in excess of the 10 CFR 100 guidelines.

Many Category V reactors do not require Seismic Category I components and systems due to the inherent safe shutdown and failsafe design of the reactor. Generally, the integrity of the reactor building at a fast burst reactor facility is not required in order to maintain doses within the applicable guidelines; therefore, the reactor building need not be Seismic Category I. The fast burst reactor core and the fuel in the storage vault should, however, be provided with appropriate Seismic Category I physical protection, if necessary, to ensure that the reactor and fuel in storage are maintained in a subcritical configuration following an earthquake. Typical seismic classification of components, systems and structures at fast burst reactor facilities is included in Table 3.2-2. Typical seismic classifications for other pulsing reactors are included in the other Design Guides, identified in Section 1.3, that provide the basic design and safety recommendations for these facilities.

3.2.2 Electric Safety Classification

Electrical components and systems required to assure safe shutdown of the reactor and/or to prevent uncontrolled release of radioactivity with radiological consequences in excess of the guidelines in Chapter 15 are designated Class IE. These components and systems are subject to many design guidelines of the Institute of Electrical and Electronics Engineers (IEEE) nuclear standards. In addition, the quality assurance requirements of 10 CFR 50, Appendix B, and ANSI N45.2 are applicable. Non-safety instrumentation, control, and electrical systems are Non-Class IE and are subject to normal industrial design and QA requirements.

Fast burst Category V reactors do not generally require Class IE electric power systems due to their inherent failsafe design upon loss of electric power. Portions of the protection system may need to be designed to Class IE standards, as described in Section 7.1.

3.2.3 Determination of System Safety Class

Components, systems, and structures are subdivided into safety classes or quality groups for the purpose of determining the applicable design and quality requirements. The terms Safety Class and Quality Group are interchangeable, with Safety Classes 1, 2 and 3 corresponding directly to Quality Groups A, B and C. Non-Safety Class (non-nuclear) components are in Quality Group D. USNRC Regulatory Guide 1.26 requires that a system be designed to a safety class if the conservatively calculated potential off-site doses due to failure of the system can exceed 0.5 rem to the whole body. If the conservatively calculated potential off-site doses due to failure of the system do not exceed 0.5 rem, the system may be designed to non-nuclear standards. Table 3.2-1 presents general guidelines for determining the minimum safety class of Category V reactor facility components, systems, and structures. Note that fast burst Category V reactors do not require Safety Class 1 components, systems, and structures. SC-1 applies to reactor coolant system pressure boundary components of commercial power reactors and some DOE reactors. Category V reactors do not have such components. In general, fast burst Category V reactors also do not require Safety Class 2 components, systems, and structures. The source terms resulting from postulated reactor accidents (see Chapter 15) will generally result in doses not exceeding the 10 CFR 100 guidelines. Components, systems, and structures of fast burst Category V reactor facilities will, therefore, generally be either Safety Class 3 or non-nuclear. Safety class guidelines for other pulsing reactors are included in the other applicable Design Guides identified in Section 1.3.

Applying the general guidelines of Table 3.2-1, Table 3.2-2 lists the likely minimum safety classes of reactor systems that are typical of fast burst Category V reactors. Table 3.2-2 is not to be interpreted as a listing of required systems for Category V reactors. It is rather a listing of systems that would be found at a hypothetical "typical facility." Individual facilities may vary substantially from this listing while still adequately protecting health and safety. Because of significant differences between Category V reactors and commercial power reactors, the Table 3.2-1 and 3.2-2 listings do not necessarily correspond to the safety class guidelines that are applicable to commercial power reactors.

The final safety class to which a component, system, or structure is actually designed is based on an evaluation of the on-site and off-site radiological consequences of postulated accidents. If the radiological consequences of all accidents are determined to be within the guidelines specified in Chapter 15, the final safety class may be the same as the minimum safety class recommended in this Design Guide. If any accidents would result in doses exceeding the guidelines in Chapter 15, some system changes may be needed. Alternatively, it may be necessary to design certain systems to a final safety class that is higher than the minimum safety class in order to increase the quality and reliability of the equipment.

As described below, determining the safety class of structures, components, and systems provides the designer with the information necessary for defining

Table 3.2-1. General Minimum Safety Class Guidelines^(1,2)

Safety Class 1 (Quality Group A)

Fast Burst Category V reactors do not generally require Safety Class 1 components, systems or structures.

Safety Class 2 (Quality Group B)

Those systems, the failure of which may result in accidents with radiological consequences in excess of the 10CFR100 guidelines.

Safety Class 3 (Quality Group C)

Those components, systems or structures that are not SC-1 or SC-2, but contain or may contain radioactive material and have the potential for causing off-site doses in excess of 0.5 rem to the whole body or the equivalent to any part of the body, but less than the 10CFR100 guidelines.

Note:

- (1) General guidance for determining the final safety class is presented in Section 3.2 and in Chapter 15.
- (2) Components, systems, and structures that contain or may contain radioactive material but are not Safety Class 1, 2 or 3 may be designed to Non-Safety Class (Non-Nuclear) standards. The maximum off-site dose resulting from the failure of a Non-Nuclear system should not exceed 0.5 rem to the whole body or the equivalent to any part of the body.

Table 3.2-2

Typical Equipment Classification for
Fast Burst Category V Reactors^(a)

System or Component	Design Guide Section	Safety Class	Quality Group	Seismic Category	10CFR50 Appendix B	Principal Design Code of Standard
Reactor Core Assembly ^(b)	4.1	-	-	I	Applies	None
Reactivity Control System ^(b)	4.2	(c)	-	I	Applies	None
Reactor Building ^(d)	6.2	(e)	-	(e)	-	ANSI A89.1, AISC Specs.
Protection System ^(f)	7.1	-	-	-	Applies	IEEE 279
Non-Safety Instrumentation and Control System	7.2	-	D	-	-	NEMA ICS-1970
Non-IE Electric System	8.2	-	D	-	-	ANSI C1-1975, NEMA MG 1-1972
Fuel Handling and Storage System	9.2	(g)	D	-	-	None
Fire Protection System	9.4	(f)	D	-	-	NFPA 802-1974
Experimental Facilities	10	(h)	(h)	(h)	(h)	None
Radioactive Waste System	11	(g)	D	-	-	None
Radiation Protection System	12	(g)	D	-	-	None

- Notes:** (a) The typical equipment classification for other pulsing reactors are included in the other Design Guide which provides the basic design and safety guidelines applicable to the particular reactor type (e.g., Category II, III, and VI Design Guides for tank-type, pool-type, and air-cooled graphite reactors, respectively).
- (b) The reactor core assembly and the reactivity control system should be designed to inherently limit reactivity bursts or pulses.
- (c) See guidelines in Section 4.2.
- (d) Generally, the reactor building need not be designed as a containment.
- (e) See guidelines in Section 6.2.
- (f) Generally for reactor operating interlocks and reactor trip function only.
- (g) Not required to be designed to safety class standards. May be designed to Non-Nuclear standards.
- (h) Experimental facilities may interface with other facility equipment (e.g., the reactor core assembly and the instrumentation, control, and electrical systems). The experimental facilities should be design to safety class, seismic category, and QA standards that are compatible with the design requirements of interfacing systems.

the codes, standards and guides that should be considered in the design of reactor facility equipment.

A. Equipment Designed to the Requirements of the ASME Boiler and Pressure Vessel Code

Fast burst Category V reactors do not generally require equipment designed to the ASME B&PV code. If a designer at a particular Category V reactor facility should determine that some systems or components should be designed to the requirements of the ASME B&PV code, the Design Guide for Category II Reactors, Light and Heavy Water Reactors, should be consulted.

B. Equipment Not Covered by the ASME Boiler and Pressure Vessel Code

Structures, components, and systems not subject to the ASME Boiler and Pressure Vessel Code may be Safety Class 1, 2 or 3 or non-nuclear. Guidelines for stress, deformation, and fatigue limits for each facility operating condition should be established and utilized by the designer. These guidelines should ensure that mechanical components are designed with sufficient conservatism to ensure the safe shutdown of the reactor and/or to prevent an uncontrolled release of radioactivity with radiological consequences in excess of the guidelines in Chapter 15.

The allowable stress, deformation, and fatigue guidelines for non-ASME code mechanical components may be based on the criteria established in applicable industry codes for similar equipment (e.g., "Manual of Steel Construction," American Institute of Steel Construction), may be derived from manufacturer's standards (e.g., "Specifications for Electric Overhead Traveling Cranes," CMAA-70, Crane Manufacturers Association of America), or may be derived from empirical methods based on field experience and /or testing. Such derived guidelines should, to the extent practical, be consistent with similar requirements for equipment in the same safety class that is designed to the ASME code.

Safety class systems, components, and structures are subject to the quality assurance requirements of 10 CFR 50, Appendix B and ANSI N45.2. Non-nuclear systems are subject to normal industrial QA practices.

3.2.4 Applicable General Design Criteria

It is recommended that the following General Design Criteria and other provisions of 10 CFR 50 and 10 CFR 100 be considered in classifying components, systems, and structures:

- A. GDC 1, "Quality Standards and Records"
- B. GDC 2, "Design Basis for Protection Against Natural Phenomena"
- C. 10 CFR 50.55a, "Codes and Standards"

D. 10 CFR 50, Appendix B, "Quality Assurance Requirements for Nuclear Power Plants and Reprocessing Plants"

E. 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants"

3.2.5 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in classifying components, systems, and structures:

A. Industry Standards and Codes

1. ANSI N18.8 (ANS 4.1), Draft, "Preparation of Design Bases for Systems that Perform Protection Functions in Nuclear Power Generating Stations"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.26, "Quality Group Classification and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants"

3.2.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 3.2.1, "Seismic Classification," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 3.2.2, "System Quality Group Classification," *ibid.*

3. Seth, W. B., "Nuclear Code Class, Safety Class, and Quality Administration: How They Tie Together," 71-PVP-56, American Society of Mechanical Engineers, 1971

B. Pending Industry Standards and Codes

None

3.2.7 Topics for Future Standards Development

None

3.3 WIND AND TORNADO LOADINGS

For the safety class systems identified in Section 3.2 that require protection against wind and tornado loadings, the following guidelines should be considered:

3.3.1 Design Guidelines

Components, systems, and structures should be designed to withstand the effects of wind and tornado loadings if they must function during and after these severe meteorological conditions in order to assure safe shutdown of a reactor and/or to prevent an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15. As a minimum, the Category V reactor core and any fuel in storage should be protected against extreme winds by a reactor building or by a fuel storage vault. Protection requirements for other facility systems outside the reactor building or vault should be based on the above general guidelines.

The characteristics of the operating basis wind speed and the design basis tornado should be based on meteorological data collected at the reactor site. Generally the value of 130 miles/hour at 30 feet above ground level equals or exceeds accepted values for the operating basis wind at nuclear power plants, except those on exposed coastal locations (see WASH 1361). The loadings resulting from the design basis tornado is likely to exceed any other wind loading. Nevertheless, there may be some structures, for example, at coastal locations, where a separate wind loading should be considered. ANSI A58.1-1972 and ASCE Paper 3269 describe two methods of determining the loading resulting from the operating basis wind speed.

3.3.2 Applicable General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criterion be considered in designing a reactor facility to withstand wind and tornado loadings:

- A. GDC 2, "Design Basis for Protection Against Natural Phenomena"

3.3.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in designing a reactor facility to withstand wind and tornado loadings:

- A. Industry Standards and Codes

- 1. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

- 2. ANSI/ANS 58.3-1977 (N182), "Physical Protection for Systems and Components Important to Safety"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"

2. USNRC Regulatory Guide 1.117, "Tornado Design Classification"

3.3.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 3.3.1, "Wind Loadings," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 3.3.2, "Tornado Loadings," ibid.

3. USNRC Branch Technical Position AAB 3-2, "Tornado Design Classification," (Appended to SRP 3.5.1.4), ibid.

4. WASH 1300, "Technical Basis for Interim Regional Tornado Criteria," USAEC, May 1974

5. WASH 1361, "Safety Related Site Parameters for Nuclear Power Plants," January 1975

6. ASCE Paper 3269, "Wind Force on Structures," Transactions of the American Society of Civil Engineers, Vol. 126, Part II (1961)

7. IAEA Safety Guide SG-S11B, Draft, "Design Basis Hurricane-Wind and Pressure Field," International Atomic Energy Agency, Vienna

B. Pending Industry Codes and Standards

1. ANSI N178 (ANS 2.3), Proposed, "Guidelines for Estimating Tornado, Hurricane, and Other Extreme Wind Parameters at Power Reactor Sites"

3.3.5 Topics for Future Standards Development

None

3.4 WATER LEVEL (FLOOD) DESIGN

For the safety class systems identified in Section 3.2 that require protection against flooding, the following guidelines should be considered.

3.4.1 Flood Design

Components, systems, and structures should be designed to withstand the effects of flooding if they must function during and after floods in order to assure safe shutdown of a reactor and/or to prevent an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15. As a minimum, the Category V reactor core and the fuel in storage should be protected against flooding in order to prevent an inadvertent criticality. Protection requirements for other facility systems outside the reactor building or vault should be based on the above general guidelines.

The magnitude of the design basis flood, including attendant wind-generated wave activity for which the reactor facility is designed, should be determined on the basis of an assessment of the flood risk from all possible causes, including seismically induced floods and dam failures. Methods of flood protection to be considered should include:

- A. Dry Site
- B. Exterior Barriers
 - 1. Levee
 - 2. Seawall or Floodwall
 - 3. Bulkhead
 - 4. Revetment
 - 5. Breakwater

3.4.2 Applicable General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criterion be considered in designing a reactor facility to withstand the effects of floods:

- A. GDC 2, "Design Basis for Protection Against Natural Phenomena"

3.4.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in designing a reactor facility to withstand the effects of floods:

- A. Industry Standards and Codes

1. ANSI N170-1976, "Standard for Determining Design Basis Flooding at Power Reactor Sites"

2. ANSI/ANS 58.3-1977 (N182), "Physical Protection for Systems and Components Important to Safety"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"

2. USNRC Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants"

3. USNRC Regulatory Guide 1.135, "Normal Water Level and Discharge at Nuclear Power Plants"

3.4.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or are distributed only on a limited basis as preliminary drafts are listed for information only

A. Background Information

1. USNRC Standard Review Plan 3.4.1, "Flood Protection," NUREG 75/087, September 1975

2. WASH 1361, "Safety-Related Site Parameters for Nuclear Power Plants," January 1975

3. IAEA Safety Guide SG-S10A, Draft, "Design Basis Floods for Nuclear Power Plants on River Sites," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

None

3.4.5 Topics for Future Standards Development

None

3.5 MISSILE PROTECTION

Components, systems, and structures that are required for assuring safe shutdown of a reactor and/or for preventing an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15 should be designed to withstand the effects of a spectrum of potential missiles, considering the probability guidelines in Section 3.1.

3.5.1 Missiles Generated at the Reactor Facility

Among the facility-generated missiles to be considered as potential hazards are those arising from failure of experiments, experimental facilities, and high-speed rotating machinery. Examples of potential missiles are explosive irradiation samples, gas cylinders, fan blades, and overhead cranes.

Equipment with a potential for generating missiles should be located such that safety class systems will not be endangered. Where the physical separation of systems does not provide adequate missile protection, barriers or operating procedures should be used.

3.5.2 Missiles Generated by Natural Phenomena

Tornados, hurricanes, earthquakes, and floods are capable of generating missiles which have the potential for disabling safety class components, systems and structures. These critical areas of a facility should be designed to withstand the effects of missiles or should be protected by structures or missile barriers. In particular, the design of the reactor building should prevent potential externally generated missiles from contacting the reactor core or fuel in the storage vault.

The design basis missiles at a particular site should be determined based on the frequency of specific severe natural phenomena and the potential consequences of missile strikes on safety class components, systems, and structures. Generally, tornado-generated missiles are considered the most limiting. A representative spectrum of tornado-generated missiles is included in WASH 1361.

3.5.3 Other Externally Generated Missiles

Aircraft crashes and explosions on nearby transportation routes may generate missiles against which safety class systems at a reactor facility should be protected. The peak overpressure resulting from an explosion may be equated to the dynamic wind pressure from the design basis tornado (see USNRC Regulatory Guide 1.91). In most cases, a facility that is designed to withstand the design basis tornado will also be protected from nearby explosion-generated missiles and overpressures. Protection, in addition to that required for the design basis tornado, should be provided when a nearby explosion may result in an overpressure that exceeds the equivalent tornado dynamic wind pressure.

3.5.4 Applicable General Design Criteria

It is recommended that the following provisions of 10 CFR be considered in designing a reactor facility to withstand the effects of missiles:

- A. GDC 2, "Design Basis for Protection Against Natural Phenomena"
- B. GDC 4, "Environmental and Missile Design Basis"

3.5.5 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in designing a reactor facility to withstand the effects of missiles:

A. Industry Standards and Codes

- 1. ANSI N177 (ANS 58.1), Draft, "Plant Design Against Missiles"
- 2. ANSI/ANS 58.3-1977 (N182), "Physical Protection for Systems and Components Important to Safety"

B. RDT Standards

None

C. Guides

- 1. USNRC Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites"

3.5.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

- 1. USNRC Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," NUREG 75/087, September 1975
- 2. USNRC Standard Review Plan 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," *ibid.*
- 3. USNRC Standard Review Plan 3.5.1.6, "Aircraft Hazards," *ibid.*
- 4. USNRC Standard Review Plan 3.5.3, "Barrier Design Procedures," *ibid.*

5. Branch Technical Position AAB 3-2, "Tornado Design Classification (Appended to SRP 3.5.1.4)," *ibid.*

6. WASH 1361, "Safety Related Site Parameters for Nuclear Power Plants," January 1975

7. Department of the Army Technical Manual TM5-1300, "Structures to Resist the Effects of Accidental Explosions," June 1969

8. IAEA Safety Guide SG-D5, Draft, "Man-Induced Events," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

1. ANS 2.12, Draft, "Guidelines for Combining Natural and External Man-Made Hazards at Nuclear Power Reactor Sites"

3.5.7 Topics for Future Standards Development

None

3.6 PROTECTION AGAINST THE DYNAMIC EFFECTS ASSOCIATED WITH POSTULATED RUPTURE OF PIPING

Fast burst Category V reactor facilities do not have high or moderate energy piping systems. Protection of components, systems, and structures against the effects of pipe ruptures is, therefore, not required.

Other Category V reactors may require protection against the dynamic effects of piping failures. Guidelines for providing this protection are included in the other applicable Design Guides identified in Section 1.3.

3.7 SEISMIC DESIGN

If it is determined from the guidelines of Section 3.2 that a component, system, or structure should be designed to Seismic Category I standards, the following seismic design guidelines are applicable.

3.7.1 Seismic Input

The seismic input to the plant design should be based on a detailed seismic investigation of the reactor site which defines the site or facility specific seismic parameters, including design response spectra and critical damping values. These parameters are discussed below. The Design Basis Earthquake (DBE) (or the Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE), if appropriate) should be defined to be consistent with the recommendations in Section 3.2.

3.7.1.1 Design Response Spectra

The design response spectra for vertical and horizontal motion may be derived from an appropriate scaling of the response spectra in USNRC Regulatory Guide 1.60. This is a relatively smooth curve derived from evaluating and statistically combining a number of response spectra derived from the records of significant earthquakes. The degree of scaling necessary is based on the relative intensity of the DBE (or the SSE, if appropriate) at the actual reactor site. The response spectra recommended in USNRC Regulatory Guide 1.60 correspond to maximum vertical and horizontal ground acceleration of 1.0 g and an accompanying displacement of 36 inches, without considering soil-structure interaction.

3.7.1.2 Critical Damping Values

The percentages of critical viscous damping used for the design and analysis of Seismic Category I structures, systems, and components should conform with or be more conservative than those recommended in USNRC Regulatory Guide 1.61, unless test data are available to support higher damping values.

3.7.2 Seismic System Analysis

General guidelines for seismic system analysis are found in USNRC Standard Review Plan 3.7.2. The seismic analysis of a facility should include the following items:

- A. Equivalent static load analysis of systems (justified based on simple fast burst Category V reactor system configuration and conservative analytical technique).
- B. Natural frequency and response load analysis, including the time-history of acceleration or response spectrum at the reactor operating location.
- C. Soil-structure interaction analysis using the finite element technique, or equivalent, where foundations are deeply embedded in soil. Shallow foundations may use finite element or multiple mass spring (shear beam) methods. Structures supported on rock may use the fixed-base approach.
- D. Floor response spectra should be developed from taking into consideration the three components of earthquake motion. The time-history method generally results in acceptable spectra. If a modal response spectra method is used, its conservatism should be verified.
- E. Appropriate methods should be used to account for torsional effects and overturning moments.
- F. The seismic analysis should account for damping in different elements of the model of a coupled system.

3.7.3 Seismic Instrumentation

Seismic instrumentation should be provided to permit the determination of the seismic response of safety class systems during an earthquake and to permit a comparison of such response with the design basis response. Such a comparison may be needed to decide whether the facility can safely continue operation following an earthquake. If the design basis response is exceeded, the reactor should be shut down. Repairs and/or inspections of safety class systems should be performed prior to returning the reactor to operation. Seismic instrumentation may be minimized if a reactor is shut down immediately upon detection of a seismic event, and is not returned to operation until safety class system integrity and operability has been verified by calculational methods, inspections, tests or repairs, as appropriate.

3.7.4 Applicable General Design Criteria

It is recommended that the following provisions of 10 CFR be considered in the seismic design of reactor facilities:

A. GDC 2, "Design Basis for Protection Against Natural Phenomena"

B. 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants"

3.7.5 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the seismic design of reactor facilities.

A. Industry Standards and Codes

1. ANSI N18.5-1974, "Earthquake Instrumentation Criteria for Nuclear Power Plants"

B. RDT Standards

The applicability of the requirements of the following RDT standard should be considered on a case basis:

1. RDT F9-2T, "Seismic Requirements for Design of Nuclear Power Plants and Test Facilities"

C. Guides

1. USNRC Regulatory Guide 1.12, "Instrumentation for Earthquakes"

2. USNRC Regulatory Guide 1.29, "Seismic Design Classification"

3. USNRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"

4. USNRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants"

5. USNRC Regulatory Guide 1.92, "Combining Modal Response and Spatial Components in Seismic Response Analysis"

6. USNRC Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components"

3.7.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 3.7.1, "Seismic Input," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 3.7.2, "Seismic System Analysis," *ibid.*

3. USNRC Standard Review Plan 3.7.3, "Seismic Subsystem Analysis," *ibid.*

4. USNRC Standard Review Plan 3.7.4, "Seismic Instrumentation," *ibid.*

5. WASH 1361, "Safety-Related Site Parameters for Nuclear Power Plants," January 1975

6. Werner, S. D., "Study of Earthquake Input Motions for Seismic Design," TID-25438, June 1970

7. Hays, W. W., et al., "Guidelines for Developing Design Earthquake Response Spectra," CERL-TR-M-114, Army Construction Engineering Research Laboratory, U. S. Geological Survey, June 1975

8. Lomenick, T. F., "Earthquakes and Nuclear Power Plant Design," ORNL-NSIC-28, Oak Ridge National Laboratory, July 1970

9. IAEA Safety Guide SG-S1, "Earthquakes and Associated Topics for Nuclear Power Plants," International Atomic Energy Agency, Vienna

10. IAEA Safety Guide SG-S2, Draft, "Seismic Analysis and Testing of Nuclear Power Plants," *ibid.*

B. Pending Industry Standards and Codes

1. ANSI N18.4 (ANS 2.1), Proposed, "Guidelines for Determining Vibratory Ground Motion for the Design Earthquake for Nuclear Facilities"

2. ANSI N643 (ANS 2.0), Proposed, "Guidelines for Retrieval, Review, Processing, and Evaluation of Records Obtained from Seismic Instrumentation"

3. ANSI 2.14, Proposed, "Determination of the Shape of Response Spectra for use in Nuclear Facilities Design"

3.7.7 Topics for Future Standards Development

None

3.8 DESIGN OF CATEGORY I STRUCTURES

If it is determined from the guidelines in Section 3.2 that a structure should be designed to Seismic Category I standards, the following design guidelines are applicable.

3.8.1 Concrete or Steel Confinement or Safety-Related Reactor Building

Steel, reinforced concrete, or prestressed concrete seismic Category I structures may be provided to prevent an uncontrolled release of radioactivity in excess of the guidelines in Chapter 15 following postulated accidents and/or to protect safety class systems from the effects of missiles and natural phenomena. Guidelines for determining the capabilities that should be incorporated in the design of the reactor building are presented in Section 6.2. The codes and standards applicable to the design of steel or concrete Seismic Category I reactor buildings are given in this section. The loads normally applicable to Seismic Category I steel or concrete reactor buildings include the following:

- A. Those loads encountered during preoperational testing.
- B. Those loads encountered during normal plant startup, operation, and shutdown.
- C. Those loads to be sustained during severe environmental conditions, including those induced by the design wind specified for the plant site.
- D. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado and the design basis earthquake (or the safe shutdown earthquake, if appropriate) specified for the plant site.
- E. Those loads to be sustained during and after abnormal plant conditions, which may include a design basis reactivity accident and the detonation of an uncontained explosive experiment.

The acceptance testing and inservice inspection of confinements and safety-related reactor buildings should include inspection throughout the life of the facility in order to verify that structural integrity, filter efficiencies, and other safety-related parameters are within limits specified for the facility.

3.8.2 Concrete and Steel Internal Structures of Steel or Concrete Confinements, or Safety-Related Reactor Buildings

Internal structures may have safety functions for which their structural integrity is important. For example, by providing support to experiments during normal operations and during seismic disturbances, internal structures should prevent the occurrence of an unplanned reactivity transient caused by movement of an experiment near the reactor.

Loads normally applicable to reactor building internal structures are similar to those considered in Subsection 3.8.1 for the reactor building.

Safety class building internal structures should be designed to permit periodic testing and inservice inspection as required to assure their continued integrity and ability to perform their safety-related functions when required.

3.8.3 Other Seismic Category I Structures

Fast burst Category V reactor facilities do not generally have other Seismic Category I structures. With the exception of the equipment in the reactor control building (which is generally remote from the reactor building) most fast burst Category V reactor systems are within the reactor building. Due to their inherent safety features, fast burst Category V reactors will fail to a safe shutdown condition upon loss of electric power and protection system signals from the reactor control building.

3.8.4 Applicable General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criteria be considered in the design of Seismic Category I reactor buildings and their internal structures. Other GDCs applicable to the design of reactor buildings are listed in Section 6.2.

- A. GDC 2, "Design Basis for Protection Against Natural Phenomena"
- B. GDC 4, "Environmental and Missile Design Bases"

3.8.5 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the design of Seismic Category I reactor buildings and their internal structures and other Seismic Category I structures. Other standards, codes, and guides applicable to the design of reactor buildings are listed in Section 6.2.

A. Industry Standards and Codes

1. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

2. ANSI A89.1-1972 (ACI 318-71), "Building Code Requirements for Reinforced Concrete"

3. ANSI A138.1-1972 (ACI 30172), "Specifications for Structural Concrete for Buildings"

4. ANSI A169.1-1972 (ACI 322-72), "Building Code Requirements for Structural Plain Concrete"

5. ACI 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures"

6. "AISC Specification for Design, Fabrication, and Erection of Structural Steel for Buildings," American Institute of Steel Construction, 1969

B. RDT Standards

The applicability of the requirements of the following RDT standard should be considered on a case basis:

1. RDT F9-2T, "Seismic Requirements for Design of Nuclear Power Plants and Test Facilities"

C. Guides

1. USNRC Regulatory Guide 1.10, "Mechanical (Cadwell) Splices in Reinforcing Bars in Category I Structures"

2. USNRC Regulatory Guide 1.15, "Testing of Reinforcing Bars for Category I Concrete Structures"

3. USNRC Regulatory Guide 1.29, "Seismic Design Classification"

4. USNRC Regulatory Guide 1.55, "Concrete Placement in Category I Structures"

5. USNRC Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (other than reactor vessels and containments)"

3.8.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 3.8.4, "Other Seismic Category I Structures," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 3.8.5, "Foundations" *ibid.*

3. Teal, E. J., "Seismic Design Practice for Steel Buildings," AISC Engineering Journal, 4th Quarter, 1975

B. Pending Industry Standards and Codes

None

3.8.7 Topics for Future Standards Development

None

3.9 MECHANICAL SYSTEMS AND COMPONENTS

The spectrum of postulated design basis events define four facility operating conditions for which corresponding stress levels should be defined. These are normal, upset, emergency, and faulted operating conditions. Table 3.9-1, describes these operating conditions and gives typical examples of events or accidents that may correspond to each operating condition. The operating condition into which a particular event may fall will depend, to some extent, on the detailed system design and reliability. Therefore, each facility should determine which postulated events and accidents correspond to each operating condition. Many factors affect the potential consequences of accidents and the frequency of their occurrence. By design, the more likely events should have lesser radiological consequences and should be accommodated within lower stress limits than the less likely accidents. A summary of consequence guidelines for each operating condition and a listing of the spectrum of postulated accidents to be considered in the design of reactor facilities are listed in Chapter 15.

3.9.1 Safety Class Mechanical Components

Due to the inherent safe shutdown and failsafe features of fast burst Category V reactors, few safety class mechanical components are generally required. Of those that are required, most are not covered by the ASME Boiler and Pressure Vessel Code (the general reference for Safety Class 1, 2 and 3 components in a commercial power reactor). If it is determined that some systems or components of a particular Category V reactor facility should be designed to meet the requirements of the ASME B&PV code, the Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors, should be consulted.

Safety class mechanical components at fast burst Category V reactors may include the following:

- A. Fuel plates.
- B. Other portions of the reactor structure (e.g., safety block and actuator).
- C. Mechanical portions of the reactivity control system.

Table 3.9-1

Facility Operating Conditions

Operating Condition	Description	Typical Design Basis Events
Normal	Events that are planned to occur regularly in the course of plant operation	Startup, shut-down, normal reactivity pulses
Upset	Incidents or occurrences of moderate frequency which may occur several times during the life of a facility	Loss of electric power, inadvertent reactor shut-down
Emergency	Infrequent incidents or occurrences which may occur during the lifetime of a facility	Reactivity burst larger than planned
Faulted	Events which are limiting faults and are not expected to occur during the life of a facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material and because they represent upper bounds on failures or accidents with a probability of occurrence sufficiently high to require a consideration in design. Generally a postulated accident is not considered in design if the probability of its occurrence resulting in unacceptable consequences is less than 10^{-7} per year	Design Basis Reactivity Accident

The allowable stress, deformation and fatigue guidelines for non-ASME code mechanical components may be based on the criteria established in applicable industry codes for similar equipment, may be derived from manufacturer's standards, or from empirical methods based on field experience and/or testing. Such derived guidelines should, to the extent practical, be consistent with similar requirements for equipment in the same safety class that is designed to the ASME code.

These guidelines should ensure that mechanical components are designed with sufficient conservatism to ensure the safe shutdown of the reactor and/or to prevent an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15.

The ASME Boiler and Pressure Vessel Code defines four service load levels (A, B, C, and D) applicable to ASME code class components based on the severity of the postulated design basis event. Table 3.9-2 summarizes the basis for the load limits applicable to service load levels A, B, C, and D. A comparable basis for non-ASME code class components should be used in defining the stress limits applicable to normal, upset, emergency, and faulted operating conditions, respectively.

Following the guidelines in USNRC Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components," the loads resulting from the Design Basis Earthquake (DBE) (or the Safe Shutdown Earthquake (SSE), if appropriate), should be superimposed on the loads resulting from faulted events in order to determine the total faulted load and the margin to the faulted operation condition load limits. The combination of seismic loads with faulted loads is postulated to ensure that the design of the reactor facility is sufficiently conservative to withstand faulted events that may be caused by the DBE (or SSE, if appropriate). As used herein, the term "withstand" implies that the accident consequences are within the corresponding consequence guidelines specified in Chapter 15.

3.9.2 Operability Tests for Safety Class Mechanical Components

The operability of all safety class mechanical components that must perform a mechanical motion in the performance of a safety function should be assured. These safety class mechanical components should be designed to perform their safety function under the environmental conditions discussed in Section 3.11. In addition, the operability of these mechanical components should be demonstrated by testing, detailed stress and deformation analyses, or by a combination of these methods.

3.9.3 Non-Safety Class Mechanical Components

Mechanical components may be designed to Non-Safety Class (non-nuclear) standards if they are not required for safe shutdown of the reactor or for preventing an uncontrolled release of radioactivity that may cause an off-site dose in excess of 0.5 rem to the whole body. Non-nuclear codes, standards, and guides are those applied to the design of normal industrial components.

Table 3.9-2

Basis for ASME Boiler and Pressure Vessel
Code Section III Service Loading Limits⁽¹⁾

Service Level	Basis
A	Level A Service Limits are those sets of limits which must be satisfied for all identified loadings to which the component or support may be subjected in the performance of its specified service function
B	Level B Service Limits are those sets of limits which must be satisfied for all identified loadings for which these service limits are designated. The component or support must withstand these loadings without damage requiring repair
C	Level C Service Limits are those sets of limits which must be satisfied for all identified loadings for which these service limits are designated. These sets of limits permit large deformations in areas of structural discontinuity. The occurrence of stress to Level C limits may necessitate the removal of the component from service for inspection or repair of damage to the component or support
D	Level D Service Limits are those sets of limits which must be satisfied for all identified loadings for which these service limits are designated. These sets of limits permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of the component from service

Note: (1) Defined in the ASME Boiler and Pressure Vessel Code, Section III, NCA 2142.2.

3.9.4 Applicable General Design Criteria

In addition to the General Design Criteria listed in the chapters describing the individual systems, it is recommended that the following provisions of 10 CFR 50 be considered in the design of mechanical systems and components:

- A. GDC 1, "Quality Standards and Records"
- B. 10 CFR 50.55a, "Codes and Standards"

3.9.5 Standards, Codes, and Guides

In addition to the standards, codes, and guides listed in the chapters describing the individual systems, it is recommended that the following be considered in the design of mechanical components:

- A. Industry Standards and Codes
 - 1. ANSI B-series standards
 - 2. "Manual of Steel Construction," American Institute of Steel Construction
- B. RDT Standards
 - None
- C. Guides
 - None

3.9.6 Supplementary Information

None

3.9.7 Topics for Future Standards Development

None

3.10 SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRIC EQUIPMENT

If it is determined from the guidelines in Section 3.2 that instrumentation and electrical equipment should be designed to Seismic Category I standards, the following qualification guidelines are applicable:

3.10.1 Seismic Qualification

Qualification of Category I instrumentation and electrical equipment generally includes tests and/or analysis to confirm operability during and after

earthquakes of magnitude up to and including the DBE (or the SSE, if appropriate). When testing is the selected method of seismic qualification, the following guidelines are applicable:

A. Equipment should be tested in the operational condition whenever practical.

B. Input excitation may be derived from single-frequency sinusoidal motions or from representative seismic motion at the equipment-mounting location.

C. Multi-axis testing is preferred, such that input motion is applied to one vertical axis and one principal horizontal axis simultaneously.

Due to the inherent safe shutdown and failsafe features of Category V reactors and their general lack of need for Class IE electric power (see Section 8.1), it is likely that only protection system I&C may need to be seismically qualified (see Section 7.1).

3.10.2 Applicable General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design criterion be considered in the seismic qualification of Category I instrumentation and electrical equipment:

A. GDC-2, "Design Basis for Protection Against Natural Phenomena"

3.10.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the seismic qualification of instrumentation and electrical equipment:

A. Industry Standards and Codes

1. IEEE 344-1975 (ANSI N41.7), "Recommended Practices for Seismic Qualification of Class IE Equipment of Nuclear Power Generating Stations"

B. RDT Standards

The applicability of the requirements of the following RDT standard should be considered on a case basis.

1. F9-1T, "Seismic Requirements for Design of Nuclear Power Plants and Test Facilities"

C. Guides

1. USNRC Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants"

3.10.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment," NUREG 75/087, September 1975

2. Branch Technical Position EICSB 10, "Electrical and Mechanical Equipment Seismic Qualification Program," *ibid.*

B. Pending Industry Standards and Codes

1. ANSI N278.2 (Series), Draft, "Requirements for Functional Qualification of Power-Driven Valve Actuators for Safety Related Applications in Nuclear Power Generating Stations"

3.10.5 Topics for Future Standards Development

None

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Safety class mechanical and fluid systems and Class IE electrical systems may be required to perform their safety functions during an accident or in the post-accident environment. These systems should be designed and qualified to be capable of performing their safety functions under all normal environmental conditions and under those accident/post-accident environmental conditions postulated to occur at the particular equipment location.

Considering the spectrum of postulated accidents at fast burst Category V reactors (see Chapter 15), there is no mechanism for large changes in the environmental conditions in the reactor building or in the fuel storage vault. As a result, no special postaccident environmental design requirements are generally imposed on Category V reactor mechanical and electrical equipment. The environmental conditions applicable to the design of safety class equipment for other pulsing reactors are described in the Design Guide which provides the basic design and safety guidelines for the particular reactor type (see Section 1.3).

If it should be determined that a particular Category V reactor facility requires qualification of mechanical and electrical equipment because of the unique accident/postaccident environmental conditions, the Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors, should be consulted.

3.12 WELDING OF NUCLEAR COMPONENTS

3.12.1 Welding Guidelines

The documents listed in the following subsections contain guidance pertinent to the welding of mechanical components at reactor facilities. Guidance pertinent to the design of these components is included in other chapters of this Design Guide.

3.12.2 Applicable General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criteria and other provisions of 10 CFR be considered in the welding of nuclear components:

- A. GDC 1, "Quality Standards and Records"
- B. 10 CFR 50, Appendix B, Section IX, "Control of Special Processes"

3.12.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the welding of components:

- A. Industry Standards and Codes
 - 1. AWS A3.0-76, "Welding Terms and Definitions"
 - 2. AWS D1.1-75, "Structural Welding Code" (with Rev. 1-76)
 - 3. AWS D10.0-69, "Standard for Qualification of Welding Procedures and Welders for Piping and Tubing"
 - 4. AWS D12.1-75, "Reinforcing Steel Welding Code"
 - 5. ANSI Z49.1-1973, "Safety in Welding and Cutting"
- B. RDT Standards

The applicability of the requirements of the following RDT standard should be considered on a case basis:

- 1. RDT F 6-6T, "Welding of Structural Components," (Supplements AWS D1.1-72)

C. Guides

None

3.12.4 Supplementary Information

None

3.12.5 Topics for Future Standards Development

None

3.13 NUCLEAR REACTOR FACILITY MATERIALS

3.13.1 Materials Guidelines

The documents listed in the following subsections contain guidance pertinent to the use of various material in components, systems or structures of a nuclear reactor facility. Appropriate quality assurance measures should be in effect to ensure the proper application of materials in reactor facility systems and to ensure the long term compatability of plant materials (see Chapter 17). Guidance pertinent to the design of these components, systems, and structures is included in other chapters.

3.13.2 Applicable General Design Criteria

It is recommended that the following General Design Criterion be considered in selection and use of nuclear reactor facility materials:

- A. GDC 1, "Quality Standards and Records"

3.13.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the selection and use of nuclear reactor facility materials:

- A. Industry Standards and Codes

- 1. ANSI N145-1973 (ASTM E 184-62), "Practice for Determining Effects of High Energy Radiation"

- 2. ANSI N640-1974 (ASTM E 419-73), "Guide for Selection of Neutron Activation Detector Materials"

- 3. ANSI N648-1975 (ASTM C 637-73), "Specification for Aggregates for Radiation Shielding Concrete"

- 4. ANSI N649-1975 (ASTM C 638-73), "Descriptive Nomenclature of Constituents of Aggregates for Radiation Shielding Concrete"

- 5. ANSI N4.1-1973 (ASTM D 2953-72), "Classification System for Polymeric Materials for Services in Ionizing Radiation"

6. ISA RP 25.1, "Materials for Instruments in Radiation Service"

B. RDT Standards

1. M 16-3T, "HEPA Filter Medium, Glass Fiber," (MIL-F-51079, with modifications and additional requirements)

2. M 17-1T, "Sodium Carbonate, Low Chloride Fire Extinguishing Agent"

C. Guides

None

3.13.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. "Building Materials Directory," Underwriters Laboratory, Inc., January 1977

2. MIL-F-51079B, "Filter Medium, Fire Resistant, High Efficiency," 1974

B. Pending Industry Standards and Codes

None

3.13.5 Topics for Future Standards Development

None

3.14 DECOMMISSIONING OF REACTOR FACILITIES

3.14.1 Decommissioning guidelines

The ERDA Manual, Chapter 0540, "Safety of AEC-Owned Reactors," requires that reactors be decommissioned in a manner that gives adequate consideration to health and safety factors. Chapter 0524, Part V, "Guidance on Maintaining Exposures to as Low as Practicable," further requires that decommissioning requirements be considered in the design of a reactor facility.

The following general guidelines apply to the decommissioning of reactor facilities and to the design and operational aspects that may affect decommissioning:

A. Three major alternatives for the retirement of a Category V reactor facility are:

1. Mothballing - The reactor is placed in protective storage. Appropriate monitoring, surveillance and security programs are required.

2. Removal of Radioactive Components and Dismanteling - All radioactive materials having activities above unrestricted use activity levels (see USNRC Regulatory Guide 1.86) should be removed from the facility. The facility is then available for unrestricted use.

3. Conversion to a New Nuclear System - Portions of a nuclear facility may be disposed of by one of the above methods. The remainder of the facility may be incorporated into a new reactor facility.

B. Reactor facility design should include provisions that will facilitate the eventual decommissioning of the facility and, in the process, will assist in maintaining occupational radiation exposure as low as reasonably achievable (see Chapter 12). Due to the relatively low radiation and surface contamination levels associated with Category V reactors after shutdown, decommissioning using contact maintenance is acceptable.

C. Procedures for decommissioning should assure that occupational and off-site radiation exposures are maintained as low as reasonably achievable (see Chapter 12).

D. Any unreviewed safety issues or changes in technical specifications necessary to accomplish decommissioning should be reviewed and approved by DOE prior to the retirement of a reactor facility.

3.14.2 Applicable General Criteria

It is recommended that the decommissioning of reactor facilities consider the guidance in 10 CFR 50.82, "Application for Termination of Licenses."

3.14.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered when decommissioning reactor facilities:

A. Industry Standards and Codes

None

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors"

3.14.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. Sande, W. E., et al., "Decontamination and Decommissioning of Nuclear Facilities - A Literature Search," Battelle Pacific Northwest Laboratories, BNWL-1975
2. AIF/NESP-009SR, "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives," Atomic Industrial Forum, November 1976
3. IAEA-179, "Decommissioning of Nuclear Facilities," International Atomic Energy Agency, Vienna, 1975

B. Pending Industry Standards and Codes

1. ANSI N550 (ANS 15.10), Proposed, "Decommissioning of Research Reactors"

3.14.5 Topics For Future Standards Development

None

REACTOR

4.1 REACTOR CORE ASSEMBLY

4.1.1 Scope

This section provides guidelines for the design of the reactor core assembly. Figure 4.1-1 is a guide to the use of this section.

4.1.2 Safety Function

- A. The reactor core assembly should maintain fuel geometry to ensure that the reactor can be safely shut down during all facility operating conditions.
- B. The reactor should inherently terminate reactivity pulses or bursts.

4.1.3 Definitions

The reactor core assembly is defined as consisting of the following components:

- A. The fuel plates (burst reactors) or fuel assemblies (other pulsing reactors).
- B. In-core reactivity control system (e.g., safety block, control rods).
- C. Reflector elements (if provided).
- D. Core support structures.
- E. In-core experimental facilities (glory hole, irradiation cell, test loop).

4.1.4 Design Guidelines

4.1.4.1 Guidelines For the Design of the Core Assembly

A. Category V reactors should be designed to inherently terminate reactivity pulses or bursts by means of a large prompt negative temperature coefficient. Once a pulse or burst is initiated by the reactivity control system (see Section 4.2), the pulse or burst characteristics should be controlled by core dynamics.

B. The reactor core reactivity control system should be designed to provide highly reproducible pulses or bursts. System design and administrative procedures should provide a high level of assurance that pulse or burst size will be no greater than the predetermined value set in by the reactivity control system.

Figure 4.1-1. Guide to the use of section 4.1.

C. Mechanical properties of Category V reactor fuel should ensure its continued integrity, especially during the repeated fast temperature transients associated with pulsing and burst operations.

D. Instrumentation should be provided to monitor reactor core power over the full range of power operation. Protection System (see Section 7.1) inputs should be provided, as required, to prevent core assembly damage in excess of the guidelines in H and I below.

E. The potential effects of experimental facilities associated with a reactor core, including sample or irradiation holes, test loops and their contents, should be considered in the design of the core assembly. The experimental facilities should be designed such that any postulated failures do not result in damage to the core assembly in excess of the guidelines in H and I below.

F. Design of mechanical components for the core assembly should consider the guidelines in Section 3.9.

G. All fuel assemblies should be clearly identified to aid proper placement in the core.

H. Fast burst reactors should consider the following additional guidelines:

1. The core should be designed to be failsafe for "off-design" operational transients (e.g., the reactor should be designed to fail in a safe shutdown condition with no fuel damage in excess of the guidelines listed below).

2. The core should be designed to reduce stresses and thermomechanical shock during super-prompt critical pulses.

3. For normal and upset facility operating conditions (defined in Table 3.9-1), burst reactors should be designed such that fuel melting and excessive fuel damage is not expected to occur during the fuel design lifetime. The reactor should be designed to facilitate a rapid return to normal operation following an upset event. Fuel damage is defined as excessive distortion or cracking of fuel plates as a direct result of reactor burst operations.

4. For emergency and faulted facility operating conditions (see Table 3.9-1), the reactor should be designed such that any fuel damage that occurs should not prevent safe reactor shutdown.

5. Increased reactor availability may be facilitated by the use of an appropriate cooling system (e.g., an N_2 system). This cooling system should not be required for safety.

6. Shrouds may be used to decouple the reactor core from external sources of low energy neutrons (e.g., neutrons scattered from ex-core experiments and from the walls of the reactor building) or from experiments located within the reactor.

I. Other pulsing reactors should consider the following additional guidelines:

1. The core should incorporate thermal-hydraulic characteristics which ensure an adequate flow of coolant during pulsing operations.

2. Fuel cladding should be designed to contain expected internal gas pressure during pulsing operations.

3. For normal and upset facility operating conditions (see Table 3.9-1) other pulsing reactors should be designed such that fuel-cladding damage or fuel melting is not expected to occur during the fuel design lifetime. The reactor should be designed to facilitate a rapid return to normal operation following an upset event. Fuel-cladding damage is defined as perforation, excessive distortion, or rupture of fuel cladding as a direct result of a particular combination of power density and coolant conditions or of fuel assembly internal gas pressure encountered during pulsing operations.

4. For emergency or faulted operating conditions (see Table 3.9-1), the reactor should be designed such that any fuel or cladding damage that occurs does not prevent safe reactor shutdown.

4.1.4.2 Maintenance, Testing, and Inspection

A. Fast burst reactor cores should be designed to permit periodic visual inspection of the fuel plates for cracks, crack propagation and other flaws that may affect reactor performance and fuel plate life. Fuel plate inspections should be conducted at routine intervals (e.g., when altering core geometry) and whenever the core behavior indicates potential fuel plate abnormalities (e.g., kinetic behavior of the core is altered, delayed critical operations do not accurately repeat, etc.).

B. Other pulsing reactors should incorporate suitable provisions for inspections or monitoring to permit a periodic assessment of the integrity of the reactor.

4.1.4.3 Interfaces With Other Systems

The reactor core assembly may interface with the following other facility systems:

- A. Reactivity Control System
- B. Protection System
- C. Fuel-Handling and Storage Systems
- D. Experimental Facilities

The design of the reactor core assembly should ensure a compatible application of codes and standards across all interfaces with other facility systems.

4.1.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs be considered in the design of a reactor core assembly:

- A. GDC 10, "Reactor Design"
- B. GDC 11, "Reactor Inherent Protection"

4.1.6 Standards, Codes, and Guides

In addition to the general standards, codes, and guides listed in Chapter 3, it is recommended that the following be considered in the design of the reactor core assembly:

- A. Industry Standards and Codes
 - 1. ANSI N18.3-1972, "Fuel Element Identification"
 - 2. ANSI N394-1975 (ANS 14.1), "Operation of Fast Pulse Reactors"
 - 3. ANSI N411-1975, "Nuclear Data Sets for Reactor Design Calculations"
 - 4. ANSI N412-1975, "The Determination of Neutron Reaction Rate Distributions and Reactivity of Nuclear Reactors"
- B. RDT Standards
 - None
- C. Guides
 - None

4.1.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

- A. Background Information
 - 1. "Fuels for Pulsed Reactors," Proceedings from the 1974 ANS Winter Meeting - Special Session on Materials Science and Technology of Pulsed Reactor Fuels, January 1976
- B. Pending Industry Standards and Codes

1. ANSI N217 (ANS 5.2), Proposed, "Standard Fission Product Yields for ^{235}U and ^{239}Pu at Thermal Fission Spectrum and 14 MeV Neutron Energies"

2. ANSI N522 (ANS 19.2), Draft, "Definitions of Reactor Physics Terms and Parameters"

3. ANS 15.13, Proposed, "Allowed Core Alterations for Research Reactors"

4.1.8 Topics for Future Standards Development

None

4.2 REACTIVITY CONTROL SYSTEM

4.2.1 Scope

This section provides guidelines for the design of a reactivity control system and for determining the minimum safety class of components of the system. Figure 4.2-1 is a guide to the use of this section.

4.2.2 Safety Function

The reactivity control system should control the reactor core reactivity to ensure that the reactor can be shut down safely and that the fuel damage guidelines in Section 4.1 are not exceeded for any facility operating condition.

4.2.3 Definition

The reactivity control system is defined as the systems that control the neutron multiplying characteristics of the reactor core in order to ensure safe operation and shutdown of the reactor. This system may include control rods, movable reflector elements, burst rods, pulsed neutron sources and other reactivity control devices.

4.2.4 Design Guidelines

4.2.4.1 Guidelines for Determining the Minimum Safety Class of the Reactivity Control System

A. Mechanical or fluid system portions of the reactivity control system should be designed to the more restrictive applicable guidelines listed below:

1. Mechanical or fluid system portions of the reactivity control system should be Safety Class 2 if they are required for ensuring the following:

a. Safe shutdown of the reactor.

b. Limited fuel damage, per the guidelines in Section 4.1.

Figure 4.2-1. Guide to the use of section 4.2.

c. Off-site radiological consequences of postulated accidents are limited to the guideline doses in 10 CFR 100.

2. Mechanical or fluid system portions of the reactivity control system should be Safety Class 3 if they are required for accomplishing 1.a or 1.b, above, and their failure will not result in off-site radiological consequences in excess of the 10 CFR 100 guidelines.

3. Mechanical or fluid system portions of the reactivity control system may be designed to non-nuclear standards if the following conditions are met:

a. Their failure does not prevent safe shutdown of the reactor.

b. Their failure does not cause damage to the reactor core assembly in excess of the guidelines in Section 4.1.

c. Their failure does not cause off-site doses in excess of 0.5 rem.

B. Electrical portions of the reactivity control system required for safety should be Class IE and should consider the guidelines for protection systems in Section 7.1. These portions need include only the parts of the reactivity control system required for ensuring safe reactor shutdown, limiting potential damage to the core assembly and for limiting potential radiological consequences of postulated accidents. Other electrical portions of the reactivity control system may be Non-Class IE.

4.2.4.2 Guidelines for the Design of a Reactivity Control System

A. A minimum of two independent and diverse reactivity control systems should be provided. These systems may utilize control rods, movable reflector elements, or other suitable means of reactivity control to provide for normal reactor operation and to ensure that the reactor can be shut down safely and fuel damage limited per the guidelines in Section 4.1.

B. The reactivity control system should be capable of accurate, rapid reactivity insertions to provide routine, reproducible reactivity pulses or bursts. This requires accurate means for determining and controlling the reactivity insertion values in order to limit the fuel temperature rise during transients. Once a pulse or burst is initiated by the reactivity control system, the transient characteristics are controlled by the dynamics of the reactor core.

C. The reactor core and reactivity control system should be designed to provide highly reproducible reactivity pulses or bursts. System design and administrative procedures should provide a high level of assurance that pulse or burst size will be no greater than the predetermined value set-in to the reactivity control system.

D. Protection system inputs to the reactivity control system should be capable of overriding any non-safety reactivity control system inputs.

E. Interlocks should be provided to preclude improper operation of the reactor in the transient mode.

F. The reactivity control system, in conjunction with the inherent, prompt negative temperature coefficient of the reactor core (see Section 4.1), should be capable of placing the reactor in safe condition following a reactivity pulse or burst.

G. The reactivity control system should be designed to "fail safe" on loss of power (electric, pneumatic, or hydraulic power, as appropriate).

H. A single failure in the reactivity control system should not prevent the system from completing its safety function when required. Application of the single-failure criterion to system design is discussed in Chapter 15.

I. The reactivity control system should accommodate upset conditions (defined in Section 3.9) within the consequence guidelines recommended in Chapter 15 and facilitate a rapid return to normal operations. General guidelines for emergency and faulted events are included in Section 3.9 and in Chapter 15.

J. Safety class portions of the reactivity control system should be Seismic Category I and qualified per the guidelines in Section 3.10.

K. The design of mechanical portions of the reactivity control system should consider the guidance in Section 3.9.

L. Burst reactors should consider the following additional guidelines:

1. Since the reactor normally operates in a "cold" condition, either of two reactivity control systems (e.g., control rods and safety block) should be capable of holding the reactor subcritical in a cold condition with the core in the most reactive experimental configuration.

2. A pulsed neutron source may be used to provide for accurately timed burst initiation.

3. Following a burst, positive means should be provided for placing the reactor in a scrammed condition. The thermomechanical shock of the burst, backed up by a scram signal from the protection system, may perform this function in some burst reactors. Other burst reactors which are not placed in a scrammed condition by the thermomechanical shock of the burst, should provide diverse positive means for ensuring reactor scram (e.g., control rod withdrawal and reflector removal).

4.2.4.3 Maintenance, Testing, and Inspection

A. The reactivity control system should be designed to permit periodic functional tests and calibrations, including tests necessary for determining the actual reactivity worth of reactivity control assemblies and experiments, to ensure that the reactivity control system will be capable of completing its safety function.

B. Testing provisions of Section 7.1 apply to portions of the reactivity control system associated with the protection system.

4.2.4.4 Interfaces With Other Systems

The reactivity control system may interface with the following other facility systems:

- A. Reactor Core Assembly
- B. Protection System
- C. Non-Safety I&C System
- D. Non-Class IE Electric Power System
- F. Experimental Facilities

The design of the reactivity control system should ensure a compatible application of codes and standards across all interfaces with other facility systems.

4.2.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs be considered in the design of the reactivity control system:

- A. GDC 10, "Reactor Design"
- B. GDC 13, "Instrumentation and Control"
- C. GDC 20, "Protection System Functions"
- D. GDC 24, "Separation of Protection and Control Functions"
- E. GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"
- F. GDC 26, "Reactivity Control System Redundancy and Capability"
- G. GDC 27, "Combined Reactivity Control Capability"

H. GDC 28, "Reactivity Limits"

I. GDC 29, "Protection Against Anticipated Operational Occurrences"

4.2.6 Standards, Codes, and Guides

A. Industry Standards and Codes

1. ANSI N394-1975 (ANS 14.1), "Operation of Fast Pulse Reactors"

B. RDT Standards

None

C. Guides

None

4.2.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

None

B. Pending Industry Standards and Codes

1. ANSI N701 (ANS 15.15), Draft, "Criteria for the Reactor Shutdown Systems of Research Reactors"

4.2.8 Topics for Future Standards Development

None

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CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED COOLANT SYSTEMS

5.1 REACTOR COOLANT SYSTEM

Fast Burst Category V reactors do not generally require a reactor coolant system. Reactor heat is usually rejected to the reactor building atmosphere. In some cases, a system used to provide an inert environment for the reactor core during operation (e.g., an N_2 system) may additionally serve to cool the reactor between pulses, thereby permitting more frequent pulsing of the reactor. This system is not required for safety.

Other Category V reactors may require a reactor coolant system. Guidelines for the design of this system are found in the other applicable Design Guides identified in Section 1.3.

5.2 SECONDARY COOLANT SYSTEM

Fast Burst Category V reactors do not require a secondary coolant system. Other Category V reactors may require a secondary coolant system. Guidelines for the design of this system are found in the other applicable Design Guides identified in Section 1.3.

5.3 RESIDUAL HEAT REMOVAL SYSTEM

Decay heat rates are generally insignificant due to the intermittent operation of the fast burst Category V reactors and their low fission product inventories. A residual heat removal system is not required to protect the fuel design limits.

CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 EMERGENCY CORE COOLING SYSTEM

Decay heat rates in fast burst reactors are generally insignificant because of the intermittent operation of the reactor and the low fission product inventory. A loss of coolant accident is a trivial accident for fast burst Category V reactors (which are usually air or nitrogen cooled). An ECCS is not required to protect the fuel design limits.

Other Category V reactors may require an ECCS. Guidelines for the design of this system are included in the other applicable Design Guides identified in Section 1.3.

6.2 REACTOR BUILDING

6.2.1 Scope

This section provides guidelines for determining the capabilities that should be incorporated in the design of the reactor building. The required capabilities determine whether the reactor building is designed as a containment, confinement, safety-related reactor building or non-safety-related reactor building. Basic guidance relevant to the design of confinements, safety-related reactor buildings and non-safety-related reactor is provided in this section. Generally, fast burst Category V reactors do not require a containment. If it should be determined that such a reactor building is required at a particular Category V reactor facility, the Category II Design Guide for Light and Heavy Water-Cooled Reactors should be consulted for containment design guidelines. Figure 6.2-1 is a guide to the use of this section.

6.2.2 Safety Function

A reactor building may have either or both of the safety functions listed below. The relationship between these safety functions and the reactor building types is illustrated in Table 6.2-1.

A. Form a barrier against the uncontrolled release of radioactivity to the environment such that the potential radiological consequences are maintained within the guidelines for the appropriate facility operating condition (see Chapter 15).

B. Provide protection against the effects of severe natural phenomena and missiles to enclosed safety class components and systems.

Figure 6.2-1. Guide to the use of section 6.2.

Figure 6.2-1. Guide to the use of section 6.2
(continued).

Table 6.2-1

Safety Functions of Various Types of Reactor Buildings

Reactor Building Type	Safety Function		Remarks
	Control Release of Radioactivity to the Environment	Protect Enclosed Systems from Natural Phenomena and Missiles	
Containment	Yes	Yes	Low-leakage, pressure retaining Seismic Category I Structure
Confinement	Yes	Yes	Directed-leakage Seismic Category I Structure (maintained at a negative pressure) with a filtered exhaust stream
Safety Related Reactor Building	No	Yes	Seismic Category I Structure
Non-safety Related Reactor Building	No	No	Industrial type building

6.2.3 Definitions

The various types of reactor buildings are defined below:

A. Containment

A containment is defined as those structures and components (including the pressure retaining containment shell and isolation valves) which enclose the reactor and associated systems and must function in order to provide a low leakage, pressure retaining barrier against the uncontrolled release of radioactivity to the environment, and which additionally protect enclosed systems against the effects of severe natural phenomena and missiles. A containment is generally not required at fast burst Category V reactor facilities.

B. Confinement

A confinement is defined as those structures and components (including the building structure, ventilation paths, filtration, and adsorption units and fans) which enclose the reactor and associated systems and must function in order to control the release of radioactivity to the environment by means of a slight negative pressure within the building, and a directed building exhaust path which provides means for reducing the quantity of radionuclides released to the environment and for enhancing atmospheric dispersion. The confinement also protects enclosed systems against the effects of severe natural phenomena and missiles.

C. Safety-Related Reactor Building

A safety-related reactor building is defined as those structures which enclose and must protect the reactor and associated systems from the effects of severe natural phenomena and missiles, but which are not intended for controlling the release of radioactivity to the environment.

D. Non-Safety-Related Reactor Building

A non-safety-related reactor building is defined as those structures which enclose the reactor and associated systems, but neither control the release of radioactivity to the environment, nor provide protection against the effects of severe natural phenomena and missiles.

6.2.4 Design Guidelines

6.2.4.1 Guidelines for Determining the Minimum Safety Class and the Capabilities of the Reactor Building

The following guidelines describe the basis for selecting a containment, confinement, safety-related reactor building or non-safety reactor building to house a reactor and associated systems. The need for a containment or confinement will be based on the potential radiological impact of the facility. The need for providing protection to enclosed systems or components against the effects of severe natural phenomena and missiles will be based on the protection

requirements of the enclosed systems. The minimum safety class guidelines for the reactor building are summarized in Table 6.2-2. Design guidelines are presented in Subsections 6.2.4.2 through 6.2.4.7.

A. Containment and Confinement

The reactor building should be designed as a containment or confinement if the radiological consequence guidelines for the appropriate facility operating condition (see Chapter 15) cannot be met without taking credit for some design feature of the reactor building in order to control the release of radioactivity to the environment. The selection of containment or confinement should be based on the degree of dose or effluent concentration reduction required. A confinement is generally capable of maintaining the radiological consequences of postulated Category V reactor accidents within the guidelines of Chapter 15. Therefore, a containment is generally not required at a fast burst Category V reactor facility. Both containment and confinement provide protection to enclosed systems and components against the effects of severe natural phenomena and missiles.

B. Safety-Related Reactor Building and Non-Safety-Related Reactor Building

Neither type is designed to control the release of radioactivity to the environment. Therefore, selection of these structures is acceptable only when the facility demonstrates an inherent degree of safety due to reduced source terms of other features that permit operation without a containment or confinement capability while maintaining the radiological consequences of postulated accidents within the guidelines for the appropriate facility operating condition (see Chapter 15).

If enclosed systems or components require the reactor building to provide protection against the effects of severe natural phenomena and missiles, the reactor building is a safety class structure and must itself withstand these hazards. A non-safety-related reactor building is equivalent to a standard industrial building and generally is not designed to withstand these hazards. Therefore, enclosed safety class systems or components should be provided with other suitable protection.

6.2.4.2 Guidelines for the Design of a Containment and Associated Systems

See the Design Guide for Category II Reactors.

6.2.4.3 Guidelines for the Design of a Confinement and Associated Systems

A. Confinement Structure and Exhaust System

1. The potential releases of radioactivity to the confinement should be considered in establishing confinement air flow rates, maximum by-pass leakage requirements for the exhaust path, filter/adsorber efficiency requirements, and other parameters of confinement design that may affect the potential off-site and on-site doses resulting from postulated accidents.

Table 6.2-2

Minimum Safety Class Recommendations for Reactor Buildings

Reactor Building Type	Potential Radiological Consequences of Accidents ⁽¹⁾	
	The radiological consequences of postulated accidents would exceed the guidelines for the appropriate facility operating condition if credit were not taken for the containment or confinement capability of the reactor building	The radiological consequences of postulated accidents would not exceed the guidelines for the appropriate facility operating condition if the building were not provided
Containment ⁽⁴⁾	SC-2 or -3 ⁽²⁾	--
Confinement	SC-2 or -3 ⁽²⁾	--
Safety-Related Reactor Building	--	SC-3 ⁽³⁾
Non-Safety Related Reactor Building	--	NN

(1) Consequence guidelines for each facility operating condition are described in Chapter 15.

(2) The safety class of the reactor building will depend on the safety class of enclosed systems. (e.g., if a reactor facility with a Safety Class 1 reactor coolant system requires a reactor building with a containment or confinement capability, that structures should be SC-2. If the reactor coolant system were SC-3, an SC-3 containment or confinement would be adequate. In general, the safety class of the reactor building need not exceed the safety class of other enclosed systems.)

(3) SC-3 based on need to provide protection to enclosed structures.

(4) Generally not required at Category V reactor facilities.

2. Filter systems installed in the confinement exhaust path should incorporate sufficient redundancy to ensure that dose guidelines for the appropriate facility operating condition (see Chapter 15) are not exceeded following a single failure in the exhaust system. Application of the single failure criterion to system design is discussed in Chapter 15.

3. The design of adsorber and filter systems in the exhaust stream should consider the guidelines of USNRC Regulatory Guides 1.52 and 1.140.

4. A single failure of a component in the confinement exhaust system should not result in back leakage of radioactivity through confinement air intakes.

5. Elevated confinement exhaust locations are preferred in order to limit on-site doses and to reduce the off-site doses by enhancing atmospheric dispersion.

6. The confinement should be a Seismic Category I structure (see Section 3.8).

7. The confinement should provide protection to enclosed systems from the effects of severe natural phenomena and missiles (see Chapter 3).

B. Confinement Isolation System

No confinement isolation system is required. The spectrum of postulated accidents (see Chapter 15) does not cause any significant overpressure in the reactor building; therefore, negligible leakage will occur through the confinement penetrations.

C. Confinement Heat Removal System

Considering the spectrum of postulated accidents (see Chapter 15), there is no mechanism for rapidly increasing pressure and temperature in the reactor building. No confinement heat removal system is required.

D. Confinement Cleanup Systems

No confinement cleanup system (a recirculating system within the confinement) is required since the dose guidelines specified in Chapter 15 can generally be met without this capability.

6.2.4.4 Guidelines for the Design of a Safety-Related Reactor Building

A. The safety-related reactor building should be a Seismic Category I structure (see Section 3.8).

B. The safety-related reactor building should provide protection to enclosed systems from the effects of severe natural phenomena and missiles (see Chapter 3).

C. No isolation system, heat removal system or cleanup system is required.

6.2.4.5 Guidelines for the Design of a Non-Safety-Related Reactor Building

A. The failure of the reactor building due to severe natural phenomena or as a consequence of postulated accidents should not cause the consequential failure of any safety class system, component or structure.

B. No isolation system, heat removal system, or cleanup system is required.

6.2.4.6 Guidelines for the Maintenance, Testing and Inspection of the Reactor Building and Associated Systems

A. The performance of periodic functional tests of the complete confinement and associated systems should be possible during periods when the reactor is shutdown.

B. Qualification testing, maintenance and in-place testing of filtration and adsorption units should meet the guidelines of USNRC Regulatory Guides 1.52 and 1.140.

6.2.4.7 Interfaces With Other Systems

The reactor building and associated systems may interface with the following other facility systems:

- A. Non-Safety I&C System
- B. Non-Class IE Electric Power System
- C. Fuel-Handling and Storage System
- D. Fire Protection System
- E. Experimental Facilities
- F. Radioactive Waste System

The design of the reactor building should ensure a compatible application of codes and standards across all interfaces with other facility systems.

6.2.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following be considered in the design of reactor buildings:

- A. Confinement

1. GDC 60, "Control of Releases of Radioactive Materials to the Environment"

2. GDC 64, "Monitoring Radioactivity Releases"

B. Safety-Related Reactor Buildings and Non-Safety-Related Reactor Buildings

None

6.2.6 Standards, Codes, and Guides

In addition to the general standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of reactor buildings:

A. Industry Standards and Codes

1. Confinements

a. ANSI A158.1-1970 (ACI 307-69), "Specification for the Design and Construction of Reinforced Concrete Chimneys"

b. ANSI/ASME N509-1976, "Nuclear Power Plant Air Cleaning Units and Components"

c. ANSI N510-1975, "Testing of Nuclear Air Cleaning Systems"

d. ANSI/UL 586-1977, "Safety Standard for Test Performance of High Efficiency Particulate Air Filter Units"

2. Safety-Related Reactor Building

None

3. Non-Safety-Related Reactor Buildings

It is recommended that the following industry standards and codes be considered in the design of a non-safety-related reactor building:

a. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

b. ANSI A89.1-1972 (ACI 318-71), "Building Code Requirements for Reinforced Concrete"

c. ANSI A138.1-1972 (ACI 301-72), "Specifications for Structural Concrete for Buildings"

d. ANSI A169.1-1972 (ACI 322-72), "Building Code Requirements for Structural Plain Concrete"

e. "Specification for Design, Fabrication and Erection of Structural Steel Buildings," American Institute of Steel Construction, 1969

f. "Code of Standard Practice for Steel Buildings and Bridges," American Institute of Steel Construction, 1976

g. ERDA Manual, Chapter 6301, "General Design Criteria"

B. RDT Standards

The applicability of the requirements of the following RDT standard should be considered on a case basis:

1. Confinements

a. E9-1T, "HEPA Filters"

2. Safety-Related Reactor Buildings and Non-Safety-Related Reactor Buildings

None

C. Guides

1. Confinements

a. USNRC Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered Safety Feature Atmospheric Cleanup System Air Filtration and Adsorption Units for Light Water-Cooled Nuclear Power Plants"

b. USNRC Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants"

c. USNRC Regulatory Guide 3.2, "Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles"

2. Safety-Related Reactor Buildings and Non-Safety-Related Reactor Buildings

None

6.2.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 6.5.1, "ESF Filter Systems," NUREG 75/087, September 1975
2. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.
3. Burchsted, C.A., et al., "National Air Cleaning Handbook," ERDA 76-21, Oak Ridge National Laboratory, March 31, 1976
4. MIL-F-51068D, "Filter, Particulate, High Efficiency, Fire Resistant"
5. "Leakage Characteristics of Openings for Reactor Housing Components," NAA-SR-MEMO-5173, Atomics International, June 20, 1960
6. Kootz, R.L., et al., "Conventional Buildings for Reactor Containment," NAA-SR-10100, Atomics International, July 25, 1965
7. Kootz, R.L., et al., "Leakage Characteristics of Conventional Building Components for Reactor Housing Construction," Trans. Am. Nuc. Soc., November 1961

B. Pending Industry Standards and Codes

1. AACC CS-8T, Draft, "Tentative Standard for High Efficiency Gas Phase Adsorber Cells," American Association of Contamination Control

6.2.8 Topics for Future Standards Development

None

6.3 ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

Potential fast burst Category V reactor operating and accident source terms will generally not expose operating personnel to airborne radiation resulting in doses in excess of the guidelines for the appropriate facility operating condition (see Chapter 15). Additionally, the accident/post-accident environmental conditions of the facility generally remain within the appropriate limits for human habitability and/or safety class equipment qualification (see Section 3.11). An ESF ventilation system is generally not required for radiation protection or environmental control at fast burst Category V reactor facilities.

If airborne toxic agents are a hazard at a particular Category V reactor facility, an ESF ventilation system may be required to protect against this hazard. The Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors, should be consulted for guidance in an ESF ventilation system design.

CHAPTER 7

INSTRUMENTATION AND CONTROL

7.1 PROTECTION SYSTEM AND SAFETY-RELATED INSTRUMENTATION

7.1.1 Scope

This section provides guidelines for the design of a protection system and safety-related display instrumentation. Figure 7.1-1 is a guide to the use of this section.

7.1.2 Safety Function

A. The protection system should initiate protective actions based on the detection of abnormal plant conditions in order to safely shut down the reactor and protect the integrity of the fuel.

B. The protection system should provide control interlocks to protect against operational errors and operation of the reactor, unless specific required conditions have been met.

C. The safety-related instrumentation should provide the facility operator with sufficient information to readily determine the status of the protection system, operate the reactor within the prescribed range, and initiate safety-related actions when necessary.

7.1.3 Definition

A. The protection system includes electrical and mechanical devices which generate a reactor trip signal and provide control interlocks (from measured process variables to the input terminals of the devices that perform the protective action).

B. Safety-related instrumentation includes displays of reactor facility parameters (generally derived from the same sensors used by the protection system) and protection system status (e.g., tripped, not tripped, interlocked, operable) that are required in order to provide the operator with sufficient information to operate the reactor within the prescribed range and to perform required manual safety functions should such actions be necessary.

7.1.4 Design Guidelines

7.1.4.1 Guidelines for Determining the Need for and Minimum Safety Class of a Protection System

A protection system should be provided if an automatically initiated reactor trip is required to assure safe reactor shutdown and to protect the fuel

Figure 7.1-1. Guide to the use of section 7.1.

integrity. Guidelines for determining consequence limits for postulated accidents are found in the following sections:

- A. Fuel damage guidelines (Section 4.1)
- B. Reactivity accident guidelines (Section 4.2)
- C. Stress to reactor core mechanical components in excess of the applicable guidelines (Section 3.9)
- D. Radiation exposure guidelines for on-site personnel (Chapter 15)
- E. Off-site radiological consequence guidelines (Chapter 15)

If a protection system is required, safety-related instrumentation should be provided to supply the facility operator with necessary information on the protection system status and on safety-related facility parameters.

Electrical portions of the protection system and safety-related instrumentation should be designed to meet electrical Class IE requirements (with the exception of power supplies, as described below). Mechanical components in the protection system are safety-related and should meet the design guidelines in Section 3.9.

If a protection system is not required, a non-safety instrumentation and control system (see Section 7.2) may adequately provide for reactor control and reactor facility system actuation.

7.1.4.2 Guidelines for the Design of the Protection System

A. The protection system should be designed to detect abnormal reactor core-related conditions that require the initiation of protective actions (see Chapter 15) in order to safely shut down the reactor and to protect the integrity of the fuel.

B. To the extent practical, the protection system inputs should be derived from signals that are direct measures of the desired reactor facility parameters (e.g., reactor power level, fuel temperature).

C. The protective actions should be automatically initiated by the protection system if insufficient time is available for manual initiation of protective actions by the facility operator.

D. All automatically initiated protective actions should be provided with a manual initiation backup that is also a part of the protection system.

E. Once initiated, a protective action should go to completion, (e.g., a protective action consisting of a sequence of individual events cannot be stopped in mid-sequence). Return to operation requires subsequent deliberate operator action.

F. The protection system should be designed to "fail safe" following a component or channel failure or loss of electric power.

G. Because of the inherent safe shutdown characteristics of Category V burst reactors, the failsafe design of the reactor, and the protection system upon loss of electric power, Class IE electric power is not required by the protection system (see Section 8.1).

H. The design of the protection system should incorporate suitable redundancy, functional diversity, or equipment diversity to assure that the protection system safety functions can be completed when required during a postulated accident with the addition of a single failure in the protection system. Application of the single failure criterion to system design is discussed in Chapter 15.

I. Redundant protection system channels should be physically separated from each other, either by suitable barriers or by distance, in order to ensure that a failure in one channel or subsystem will not cause a failure in another protection system channel or subsystem.

J. The protection system should be electrically separated (by means of buffer amplifiers or other suitable, qualified isolation devices) from non-safety instrumentation and control systems which may receive protection system output signals, or which may be interlocked by the protection system. This electrical separation should ensure that a failure of the non-safety I&C system will not degrade the ability of the protection system to perform its safety functions when required. Isolation devices at the protection system boundary are considered to be part of the protection system.

K. The reactor facility parameters monitored by the protection system, and the status of the protection system itself, should be displayed on safety-related instrumentation which provides information to the facility operator such that the following items can be readily determined:

1. The current status of the protection system.
2. The current values of safety-related facility parameters monitored by the protection system.
3. The need for safety-related operator actions during accident and post-accident conditions.

L. Protection system instrumentation and controls sufficient for placing the reactor in a safe shutdown condition should be provided in a location remote from the control room for use during any condition which may make the control room uninhabitable when the reactor is in a startup or operating mode. As a minimum, this should include the ability to manually scram the reactor from a remote location.

M. Protection system instrument trip setpoints should be established with sufficient margin between the technical specification safety limit for the

process variable and the nominal trip setpoint so that automatic protective actions will correct an abnormal situation before a safety limit is exceeded. This margin should allow for the following:

1. Inaccuracy of the instrument
2. Uncertainties in calibration
3. Instrument drift which occurs in the interval between calibration
4. Instrument response time.

Note, however, that the protection system need not respond fast enough to limit the size of a prompt burst reactivity transient. The burst is inherently limited by the prompt-negative temperature coefficient characteristics of the reactor core (see Section 4.1). For burst operations, the protection system should be designed to limit potential post-burst consequences by placing the reactor core in a scrammed condition. Some Category V reactors may have an additional inherent safety feature to ensure post-burst shutdown of the reactor by dropping the safety block from the core due to the thermomechanical shock of larger bursts (see Section 4.2).

N. The protection system should be protected from the effects of severe natural phenomena and missiles as described in Chapter 3.

7.1.4.3 Maintenance, Testing, and Inspection

A. Means should be provided for checking the operational availability of protection system sensors.

B. Means should be provided for the periodic testing (generally before reactor startup) and calibration of protection system instrument channels and interlocks.

C. The reactor operator should be provided with displays to indicate if protection system channels have been bypassed or removed from operation during maintenance, testing, or inspection if such activities are performed during reactor operation.

7.1.4.4 Interfaces With Other Systems

The protection system may interface with the following other facility systems:

- A. Reactor Core Assembly
- B. Reactivity Control System
- C. Reactor Building
- D. Non-Safety I&C System

E. Experimental Facilities

The design of the protection system should ensure a compatible application of codes and standards across all interfaces with other facility systems.

7.1.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs and other provisions of 10 CFR 50 be considered in the design of a protection system:

- A. GDC 13, "Instrumentation and Controls"
- B. GDC 19, "Control Room"
- C. GDC 20, "Protection System Functions"
- D. GDC 21, "Protection System Reliability and Testability"
- E. GDC 22, "Protection System Independence"
- F. GDC 23, "Protection System Failure Modes"
- G. GDC 24, "Separation of Protection and Control Systems"
- H. GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"
- I. GDC 29, "Protection Against Anticipated Operational Occurrence"
- J. 10 CFR 50.55a, "Codes and Standards"

7.1.6 Standards, Codes, and Guides

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of a protection system:

- A. Industry Standards and Codes
 - 1. IEEE 279-1971 (ANSI N42.7-1972), "Criteria for Protection Systems for Nuclear Power Generating Stations"
 - 2. IEEE 336-1977 (ANSI N45.2), "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
 - 3. IEEE 338-1975, "Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems"

4. IEEE 352-1975 (ANSI N41.4-1976), "General Principles for Reliability Analysis of Nuclear Power Generating Stations"

5. IEEE 380-1975, "Definition of Terms Used in IEEE Nuclear Power Generating Station Standards"

6. IEEE 384-1977 (ANSI N41.14), "Criteria for Independence of Class IE Equipment and Circuits"

7. IEEE 420-1973 (ANSI N41.17), "Class IE Control Switchboards for Nuclear Power Generating Stations"

8. IEEE 494-1974, "Method for Identification of Documents Related to Class IE Equipment and Subsystems for Nuclear Power Generating Stations"

9. IEEE 500-1977, "Guide to the Collection and Presentation of Electrical, Electronic and Sensing Component Reliability Data for Nuclear Power Generating Stations"

10. IEEE 566-1977, "Recommended Practice for Design of Display and Control Facilities for Control Rooms of Nuclear Power Generating Stations"

11. IEEE 577-1976, "Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Power Generating Stations"

12. ANSI N544-1968, "Signal Connectors for Nuclear Instruments"

13. ANSI/ANS 59.3-1977, "Safety Related Control Air Systems"

B. RDT Standards

The applicability of the requirements of the following RDT standards should be considered on a case basis:

1. C1-1T, "Instrumentation and Control Equipment Grounding and Shielding Practices"

2. C16-1T, "Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems"

3. C16-2T, "Protection System Logic"

4. C16-3T, "PPS Buffers"

5. C16-4T, "Protection System Comparator"

C. Guides

1. USNRC Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Function"

2. USNRC Regulatory Guide 1.47, "By-Passed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"

3. USNRC Regulatory Guide 1.62, "Manual Initiation of Protective Action"

4. USNRC Regulatory Guide 1.75, "Physical Independence of Electrical Systems"

5. USNRC Regulatory Guide 1.105, "Instrument Spans and Setpoints"

6. USNRC Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems"

7.1.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 7.2, "Reactor Trip System," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 7.5, "Safety-Related Display Instrumentation," *ibid.*

3. USNRC Standard Review Plan 7.6, "All Other Instrumentation Required for Safety," *ibid.*

4. Branch Technical Position EICSB 21, "Guidance for Application of Regulatory Guide 1.47," *ibid.*

5. Branch Technical Position EICSB 22, "Guidance for Application of Regulatory Guide 1.22," *ibid.*

6. Branch Technical Position EISCB 24, "Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times," *ibid.*

7. Branch Technical Position EISCB 26, "Requirements for Reactor Protection System Anticipatory Trips," *ibid.*

8. Hanauer, S.H., and C.S. Waler, "Design Principles of Reactor Protection Systems," ORNL-NSIC-51, Oak Ridge National Laboratory, September 1968

9. Hagan, E.W., "Reactor Protection Systems: Philosophies and Instrumentation Reviews from Nuclear Safety," ORNL-NSIC-111, Oak Ridge National Laboratory, July 1973

10. Harrer, J.M., and Beckerly, J.G., "Nuclear Power Reactor Instrumentation Systems Handbook," Volumes 1 and 2, TID-25952-P1 and P2, USAEC 1973 and 1974

11. IAEA Safety Guide SG-D3, Draft, "Protection Systems in Nuclear Power Plants," International Atomic Energy Agency, Vienna

12. IAEA Safety Guide SG-D8, "Instrumentation and Control," ibid.

13. "Standards and Practices for Instrumentation" Fifth Edition, Instrument Society of America, 1977

B. Pending Industry Standards and Codes

1. IEEE P603, Draft, "Proposed Trial-Use Standard Criteria for Safety Systems for Nuclear Power Generating Stations"

2. ANSI N220 (ANS 4.3.1), Proposed, "Functional Classification of Digital Computers in Nuclear Power Generating Stations"

3. ANSI N701 (ANS 15.15), Draft, "Criteria for the Reactor Shutdown Systems of Research Reactors"

4. ANS 4.3.2, Proposed, "Criteria for the Application of Digital Computers in Protection Systems for Nuclear Power Generating Stations"

7.1.8 Topics for Future Standard Development

None

7.2 NON-SAFETY INSTRUMENTATION AND CONTROL SYSTEMS

7.2.1 Scope

This section provides guidelines for the design of non-safety instrumentation and control systems. Figure 7.2-1 is a guide to the use of this section.

7.2.2 Safety Function

Non-safety instrumentation and control (I&C) systems have no safety function unless they interface with the protection system. In that event, the operation or failure of the non-safety I&C must not prevent the interfacing protection system from completing its safety function when required.

7.2.3 Definition

Non-safety instrumentation and control systems include all I&C that is not a part of the protection system or associated safety-related instrumentation.

Figure 7.2-1. Guide to the use of section 7.2.

7.2.4 Design Guidelines

7.2.4.1 Guidelines for the Design of Non-Safety Instrumentation and Control (I&C) Systems

- A. Non-safety I&C systems may receive outputs from the protection system.
- B. Non-safety I&C systems may not provide inputs to a protection system.
- C. Any interconnections with a protection system should be designed to provide adequate electrical separation and independence between the two systems (e.g., via protection system buffer amplifiers or other suitable isolation devices that are part of the protection system), so that the operation or failure of the non-safety class I&C system will not degrade the ability of the protection system to perform its safety function when required.
- D. Physical separation or protection with barriers should be utilized as a means of assuring that failures in a non-safety I&C system will not prevent a protection system from completing its safety function when required.
- E. Non-safety I&C systems need not be qualified for the environmental conditions resulting from severe natural phenomena or postulated accidents.

7.2.4.2 Maintenance, Testing, and Inspection

Non-safety I&C systems should be designed to facilitate periodic testing and calibration to assure normal reactor operation within appropriate facility operating limits.

7.2.4.3 Interfaces With Other Systems

The non-safety I&C systems may interface with the following other facility systems:

- A. Reactivity Control System
- B. Reactor Building
- C. Protection System
- D. Non-Class IE Electric Power System
- E. Fire Protection System
- F. Experimental Facilities
- G. Radioactive Waste System

The design of the non-safety I&C systems should ensure a compatible application of codes and standards across all interfaces with other facility systems.

7.2.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs be considered in the design of non-safety instrumentation and control systems:

- A. GDC 13, "Instrumentation and Control"
- B. GDC 24, "Separation of Protection and Control Systems"

7.2.6 Standards, Codes, and Guides

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of non-safety instrumentation and control systems:

A. Industrial Standards and Codes

- 1. ANSI C1-1975 (NFPA 70-1975), "National Electrical Code"
- 2. ANSI C2-1977, "National Electrical Safety Code"
- 3. ANSI N413-1974, "Guidelines for the Documentation of Digital Computer Programs"
- 4. IEEE 384-1977 (ANSI N41.14), "Criteria for Independence of Class IE Equipment and Circuits"
- 5. ISA, "Standards and Practices for Instrumentation," 5th Edition
- 6. NEMA IS 1.1, "Enclosures for Industrial Controls"
- 7. NEMA ICS-1970, "NEMA Standards Publication, Industrial Controls and Systems"
- 8. ANS STD 3-1971, "Recommended Programming Practices to Facilitate the Interchange of Digital Computer Programs"

B. RDT Standards

The applicability of the requirements of the following RDT standards should be considered on a case basis:

- 1. C1-1T, "Instrumentation and Control Equipment Grounding and Shielding Practices"
- 2. C17-4T, "General Instrumentation"
- 3. C17-8T, "Annunciators for Control Systems"

C. Guides

1. USNRC Regulatory Guide 1.75, "Physical Independence of Electric Systems"

7.2.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 7.7, "Control Systems Not Required for Safety," NUREG 75/087, September 1975
2. Harrer, J.M. and Beckerly, J.G., "Nuclear Power Reactor Instrumentation System Handbook," Volumes 1 and 2, TID-25952-P1 and -P2, USAEC, 1973 and 1974
3. "Standards and Practices for Instrumentation" Fifth Edition, Instrument Society of America, 1977

B. Pending Industry Standards and Codes

1. ANS 4.3.4, Proposed, "Criteria for the Application of Digital Computers in Non-Safety-Related Functions for Nuclear Power Generating Stations"

7.2.8 Topics for Future Standards Development

None

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CHAPTER 8

ELECTRIC POWER

8.1 CLASS IE ELECTRIC POWER SYSTEMS

Category V reactor facilities do not generally require a Class IE electric power system. This is justified by the following general characteristics of these reactor facilities upon loss of all electric power:

A. The protection system and any safety class mechanical systems fail safe and place the reactor in a safe shutdown condition.

B. No active, electric-powered systems are required for shutdown cooling of the reactor.

C. No active, electric-powered systems are required for preventing an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15.

If it should be determined that a Class IE electric power system is required at a particular Category V reactor facility, reference should be made to the Design Guide for Category II Reactors, Light and Heavy Water Cooled, Reactors, for design guidelines.

8.2 NON-CLASS IE ELECTRIC POWER SYSTEMS

8.2.1 Scope

This section provides guidance relevant to the design of a Non-Class IE electric power system. Figure 8.2.1 is a guide to the use of this section.

8.2.2 Safety Function

The Non-Class IE electric power system has no safety function unless it interfaces with a Class IE electric power system. In that event, operation or failure of the Non-Class IE electric power system must not prevent the interfacing Class IE system from completing its safety function when required.

8.2.3 Definition

The Non-Class IE electric power system is the non-safety-related portion of the facility electric power system. It includes the off-site power supply and non-safety-related portions of the on-site electric generation and distribution systems.

Figure 8.2-1. Guide to the use of section 8.2.

8.2.4 Design Guidelines

8.2.4.1 Guidelines for the Design of a Non-Class IE Electric Power System

A. Any interconnections between a Non-Class IE electric power system and a Class IE electric power system should be designed to provide adequate electrical separation and independence between the two systems.

B. Physical separation or protection with barriers should be utilized as a means of assuring that failures in a Non-Class IE system will not prevent a Class IE system from completing its safety function when required.

C. Non-Class IE electric systems need not be qualified for the environmental conditions resulting from severe natural phenomena or postulated accidents.

8.2.4.2 Maintenance, Testing, and Inspection

Interconnections between Class IE systems (if provided) and Non-Class IE systems should be testable to verify that Non-Class IE loads or subsystems can be isolated from the Class IE system when necessary (e.g., via circuit breakers).

8.2.4.3 Interfaces With Other Systems

The Non-Class IE electric power system may interface with the following other facility systems:

- A. Reactivity Control System
- B. Non-Safety Instrumentation and Control Systems
- C. Fire Protection System
- D. Experimental Facilities
- E. Radioactive Waste System
- F. Radiation Protection System

The design of the Non-Class IE electric power system should ensure a compatible application of codes and standards across all interfaces with other facility systems.

8.2.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDC be considered in the design of Non-Class IE electric power systems:

- A. GDC 17, "Electric Power Systems"

8.2.6 Standards, Codes, and Guides

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of Non-Class IE electric power systems:

A. Industry Standards and Codes

1. ANSI C1-1975 (NFPA 70-1975), "National Electrical Code"
2. ANSI C2-1977, "National Electrical Safety Code"
3. IEEE 27-1969 (ANSI C37.10-1969), "Switchgear Assemblies, Including Metal Enclosed Bus"
4. IEEE 141-1976, "Electric Power Distribution for Industrial Plants"
5. IEEE 142-1972 (ANSI C114.1-1973), "Recommended Practice for Grounding Industrial and Commercial Power Systems"

B. RDT Standards

None

C. Guides

None

8.2.7 Supplementary Information

None

8.2.8 Topics for Future Standards Development

None

CHAPTER 9

AUXILIARY SYSTEMS

9.1 COOLING WATER SYSTEMS AND THE ULTIMATE HEAT SINK

Fast burst Category V reactors do not require cooling water systems. Heat from the reactor core is rejected to the reactor building atmosphere. The reactor building may utilize a ventilation system (as described in Section 6.2) to transfer heat to the outside environment.

Other Category V reactors may require cooling water systems. Guidelines for the design of these systems are included in the other applicable Design Guides identified in Section 1.3.

9.2 FUEL-HANDLING AND STORAGE SYSTEM

9.2.1 Scope

This section provides guidelines for the design of a fuel-handling and storage system. The ERDA Manual, Chapter 0530, "Nuclear Criticality Safety," provides guidelines for criticality safety in fuel-handling and storage systems. New or spent fuel to be transported from a reactor facility for reprocessing, storage, or disposal should be prepared for shipment in accordance with the provisions of the ERDA Manual, Chapter 0529, "Safety Standards for the Packaging of Fissile Material and Other Radioactive Material." Figure 9.2-1 is a guide to the use of this section.

9.2.2 Safety Function

The fuel-handling and storage system should provide for handling and storage of new and irradiated fuel plates and an assembled reactor in a scrambled condition in a manner which ensures there will be no inadvertent criticality.

9.2.3 Definition

The fuel-handling and storage system may include those components, systems, and structures required for performing the following functions:

- A. Storage of fuel plates, control rods, burst rod, safety block, and any other parts of the reactor core or experiments which may contain fissile material.
- B. Handling fuel during movement between storage areas and the reactor operating area, including core assembly and disassembly.
- C. Storage of an assembled reactor core in a scrambled condition when the reactor is not in use.

Figure 9.2-1. Guide to the use of section 9.2.

9.2.4 Design Guidelines

9.2.4.1 Guidelines for Determining the Minimum Safety Class of Fuel-Handling and Storage System Components

A. Components of a fuel-handling and storage system should be a minimum of Safety Class 3 if:

1. They are required to maintain fuel in a subcritical configuration.
2. Their failure may result in potential off-site doses in excess of 0.5 rem.

B. Other portions of the fuel-handling and storage system may be non-nuclear.

9.2.4.2 Guidelines for the Design of Fuel Handling and Storage Systems

A. Since most fuel-handling is performed manually, procedures should be in effect during fuel-handling to ensure that subcritical conditions are maintained in the reactor core and in the fuel storage vault.

B. Tools used for handling fuel should be designed to minimize the likelihood of dropping the fuel.

C. Contact maintenance procedures used during fuel-handling should minimize the spread of any surface contamination on fuel or other reactor parts.

D. The design of mechanical portions of the fuel-handling and storage system should consider the guidance in Section 3.9.

E. Fuel storage facilities should be designed to maintain the arrays of fuel in a subcritical configuration (e.g., K_{eff} less than approximately 0.95) when fully loaded with fuel.

F. Fuel storage vaults should include provisions for drainage.

G. Safety class portions of the fuel storage system should be Seismic Category I.

H. As necessary, to ensure that the consequences of handling accidents will not exceed the guidelines in Chapter 15, the reactor facility and the fuel/equipment handling system should be designed to prevent damage to the reactor core, fuel plates or safety class equipment if a heavy load is dropped or mishandled. The facility design may afford this protection by providing physical separation or protection of the reactor core and safety class equipment by suitable structures. An alternative is to design the handling system such that a connected load will not drop in the event of a single-failure or malfunction (e.g. handling system design incorporates multiple, independent load paths).

9.2.4.3 Maintenance, Testing, and Inspection

None

9.2.4.4 Interfaces With Other Systems

The fuel-handling and storage system may interface with the following other facility systems:

- A. Reactor Core Assembly
- B. Reactivity Control System
- C. Reactor Building
- D. Fire Protection System
- E. Radioactive Waste Systems
- F. Radiation Protection System

The design of the fuel-handling and storage system should ensure a compatible application of codes and standards across all interfaces with other facility systems.

9.2.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs be considered in the design of fuel-handling and storage system components:

- A. GDC 61, "Fuel Storage and Handling and Radioactivity Control"
- B. GDC 62, "Prevention of Criticality in Fuel Storage and Handling"
- C. GDC 63, "Monitoring Fuel and Waste Storage"

9.2.6 Standards, Codes, and Guides

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following also be considered in the design of fuel-handling and storage systems:

- A. Industry Codes and Standards
 - 1. ANSI B30.11-1973, "Safety Standard for Monorail Systems and Underhung Cranes"
 - 2. ANSI B30.2.0-1976, "Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)"

3. ANSI B30.16-1973, "Safety Standard for Overhead Hoists"

4. ANSI N16.1-1975, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"

5. ANSI N16.5-1975, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials"

6. ANSI N16.9-1975, "Validation of Computational Methods for Nuclear Criticality Safety"

7. CMAA Specification No. 70, "Specification for Electric Overhead Traveling Cranes," Crane Manufacturers Association of America

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"

2. USNRC Regulatory Guide 3.41, "Validation of Computational Methods for Nuclear Criticality Safety"

3. USNRC Regulatory Guide 3.43, "Nuclear Criticality Safety in the Storage of Fissile Materials"

9.2.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plant 9.1.1, "New Fuel Storage," NUREG 75/087, September 1975

B. Pending Industry Standards and Codes

None

9.2.8 Topics for Future Standards Development

None

9.3 PURIFICATION AND MAKEUP WATER SYSTEMS

Fast burst Category V reactors do not require purification and makeup water systems. As described in Sections 5.1, 5.2 and 9.1, a fast burst Category V reactor does not utilize water cooling systems.

Other Category V reactors may require a purification and makeup water system. Guidelines for the design of this system are included in the other applicable Design Guides identified in Section 1.3.

9.4 FIRE PROTECTION SYSTEM

9.4.1 Scope

General requirements for fire protection systems are specified in the ERDA Manual, Chapter 0552, "Industrial Fire Protection." This section provides additional guidelines for the design of fire protection systems that minimize the adverse effects of fire on safety class structures, systems, and components. Figure 9.4-1 is a guide to the use of this section.

9.4.2 Safety Function

The fire protection system should assure that the effects of a fire or a combustion explosion do not prevent other components, systems, or structures from completing their safety functions when required.

9.4.3 Definition

In its broadest context, a fire protection system is a comprehensive program that includes all aspects of fire prevention, detection, extinguishing, and confinement that are necessary to minimize the effects of a fire at a reactor facility. The fire protection system encompasses fire prevention, detection, annunciation, control, confinement, suppression, and extinguishing in addition to administrative procedures, fire brigade organization, inspection, maintenance, training, quality assurance, and testing.

9.4.4 Design Guidelines

9.4.4.1 Guidelines for Determining the Minimum Safety Class of Fire Protection System Equipment

A. Portions of the fire protection system should be designed to a minimum of Safety Class 3 if their failure may prevent safe reactor shutdown, cause an inadvertent criticality, or result in potential off-site doses in excess of 0.5 rem.

B. Portions of the fire protection system passing through safety class structures (e.g., reactor building) should be designed to a safety class compatible with the structure.

Figure 9.4-1. Guide to the use of section 9.4.

C. Other portions of the fire protection system may be non-nuclear.

9.4.4.2 Guidelines for the Design of a Fire Protection System

A. The fire protection system requires the integration of component parts encompassing the following three basic levels of protection:

1. Prevention of fires and combustion explosions (includes fire prevention procedures, facility design, material selection, and quality assurance requirements).

2. Rapid detection, annunciation, and extinguishing of fires that are not prevented (includes fire or smoke detection, alarm and suppression systems that operate manually or automatically).

3. Measures to minimize the effects of fires that are not promptly detected and extinguished (includes confinement and separation features, inert gas systems, and fire-resistant designs).

B. Extinguishing systems used within the reactor building or fuel storage vault should not result in flooding the reactor, fuel in storage or engineered safety features. Flooding the reactor or the fuel in storage could produce an inadvertent criticality. Flooding engineered safety features, such as the reactor building ventilation filters, could result in reduced safety system effectiveness and higher radiological consequences of accidents.

C. Total reliance should not be placed on a single fire suppression system. Appropriate backup capability should be provided.

D. The design of mechanical portions of the fire protection system should consider the guidance in Section 3.9.

E. A single failure in the fire suppression system should not impair both the primary and backup fire suppression capability. Application of the single-failure criterion to system design is discussed in Chapter 15.

F. The failure or inadvertent operation of the fire suppression system should not impair the safety capacity of structures, systems, and components.

G. Fire protection systems for fast burst Category V reactors do not generally require the capability to protect against sodium fires. If it should be determined that a particular Category V reactor facility requires a sodium fire protection system, the applicable guidelines should be consulted in the Design Guide for Category IV (Liquid-Metal Cooled) Reactors.

H. Administrative procedures and facility design reviews should assure that:

1. Noncombustible and fire resistant materials are used wherever practical in the facility.

2. The use of combustible materials is controlled.

3. Work involving ignition surces (e.g., welding) is conducted under closely controlled conditions.

4. Proper housekeeping requirements are delineated to minimize accumulation of unwanted combustibles.

5. Use of water to fight fires in the reactor building or the fuel storage vault is closely supervised by qualified individuals in order to prevent an inadvertent criticality.

9.4.4.3 Maintenance, Testing, and Inspection

Fire protection system equipment should be designed to permit periodic maintenance, testing, and inspection to assure the operability of system hardware.

9.4.4.4 Interfaces With Other Systems

The fire protection system may interface with the following other facility systems.

- A. Reactor Building
- B. Non-Safety Instrumentation and Control System
- C. Non-Class IE Electric Power System
- D. Fuel-Handling and Storage System
- E. Radioactive Waste System

The design of the fire protection system should assure a compatible application of codes and standards across all interfaces with other facility systems.

9.4.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDC be considered in the design of a fire protection system:

- A. GDC 3, "Fire Protection"

9.4.6 Standards, Codes, and Guides

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of a fire protection system:

- A. Industry Codes and Standards

1. Fire Protection Equipment

a. ANSI A9.1-1974 (NFPA 101-1973), "Code for Safety to Life from Fire in Buildings and Structures"

b. ANSI A54.1-1973 (NFPA 12-1973), "Carbon Dioxide Extinguishing System"

c. ANSI A54.2-1974 (NFPA 15-1973), "Water Spray Systems for Fire Protection"

d. ANSI A54.3-1973 (NFPA 13-1973), "Installation of Sprinkler Systems"

e. ANSI A54.4-1974 (NFPA 12A-1973), "Halogenated Fire Extinguishing Agent Systems - Halon 1301"

f. ANSI A54.5-1974 (NFPA 12B-1973), "Halogenated Fire Extinguishing Agent Systems - Halon 1211"

g. ANSI Z112.1-1974 (NFPA 10-1974), "Installation, Maintenance and Use of Portable Fire Extinguishers"

h. ANSI Z273.1-1974 (NFPA 14-1974), "Installation of Standpipe and Hose Systems"

i. ANSI Z274.1-1974 (NFPA 16-1974), "Installation of Foam-Water Sprinkler Systems and Foam-Water Spray Systems"

j. ANSI Z275.1-1974 (NFPA 17-1973), "Dry Chemical Extinguishing Systems"

k. ANSI Z285.1-1974 (NFPA 69-1973), "Explosion Prevention Systems"

l. ANSI Z286.1-1974 (NFPA 11-1973), "Foam Extinguishing Systems"

m. ANSI Z286.2-1974 (NFPA 11A-1970), "High Expansion Foam Systems"

n. ANSI Z286.3-1974 (NFPA 11B-1973), "Synthetic Foam and Combined Agent Systems"

o. NFPA 13A-1971, "Maintenance of Sprinkler Systems"

p. NFPA 20-1974, "Standard for the Installation of Centrifugal Fire Pumps"

q. NFPA 22-1974, "Water Tanks"

r. NFPA 70B-1975, "Electrical Equipment Maintenance"

s. NFPA 72D-1975, "Standard for the Installation, Maintenance and Use of Proprietary Protective Signaling Systems"

t. NFPA 72E-1974, "Automatic Fire Detectors"

u. NFPA 75-1972, "Electronic Computer/Data Processing Equipment"

v. NFPA 80-1975, "Fire Doors and Windows"

w. NFPA 90A-1975, "Air Conditioning and Ventilating Systems"

2. Fire Protection Administration

a. NFPA 6-1974, "Industrial Fire Loss Prevention"

b. NFPA 7-1974, "Fire Emergencies Management"

c. NFPA 8-1974, "Effects of Fire on Operations Management Responsibility"

d. NFPA 24-1973, "Standard for Outside Protection"

e. NFPA 26-1958, "Supervision of Valves"

f. NFPA 27-1975, "Private Fire Brigade"

g. NFPA 30-1973, "Flammable and Combustible Liquids Code"

h. NFPA 51B-1971, "Cutting and Welding Processes"

i. NFPA 241-1975, "Safeguarding Buildings During Construction and Demolition Operations"

j. NFPA 251-1972 (ASTM E 119-1971), "Standard Methods of Fire Tests of Building Construction and Materials"

k. NFPA 255-1972 (ASTM E 84-1971), "Method of Test of Surface Burning Characteristics of Building Material"

l. NFPA 601-1975, "Guard Service in Fire Loss Prevention"

m. NFPA 801-1975, "Recommended Fire Protection Practice for Facilities Handling Radioactive Materials"

n. NFPA 802-1974, "Recommended Fire Protection Practice for Nuclear Reactors"

B. RDT Standards

None

C. Guides

1. Fire Protection System Equipment

a. USNRC Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants"

2. Fire Protection System Administration

a. USNRC Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants"

b. USNRC Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants"

9.4.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 9.5.1, "Fire Protection System" Rev. 1, NUREG 75/087, September 1975

2. Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," (attached to SRP 9.5.1, Rev. 1), *ibid.*

3. NUREG-0050, "Recommendations Related to Browns Ferry Fire," USNRC, February 1976

4. IAEA Safety Guide SG-D2, Draft, "Safety Guide for Fire Protection in Nuclear Power Plants" International Atomic Energy Agency, Vienna

5. WASH 1245-1, "Standard for Fire Protection of AEC Electric Computer/Data Processing Systems"

B. Pending Industry Standards and Codes

1. ANS 15.17, Proposed, "Fire Protection for Research Reactors"

2. ANSI N18.10/ANS 59.4, Draft, "Generic Requirements for Light Water Nuclear Power Plant Fire Protection"

3. IEEE P634, Draft, "Standard for Cable Penetration Fire Stop Qualification Test"

9.4.8 Topics for Future Standards Development

None

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CHAPTER 10

EXPERIMENTAL FACILITIES

10.1 SCOPE

This section provides guidelines for the design of experimental facilities, including those that are a part of the reactor core (e.g., irradiation holes). The ERDA Manual, Chapter 0530, "Nuclear Criticality Safety," provides guidelines for criticality safety in experimental facilities utilizing fissionable material. In addition, Chapter 0531, "Safety of Non-Reactor Nuclear Facilities," may be applicable to some portions of experimental facilities that are not associated with reactor systems. Figure 10.1-1 is a guide to the use of this chapter.

10.2 SAFETY FUNCTION

The safety functions of the experimental facilities are those associated with potential interfaces with the reactor core assembly (see Section 4.1), the reactivity control system (see Section 4.2), or other safety class systems.

Experimental facilities which do not interface with safety class systems generally do not have a safety function.

10.3 DEFINITION

Experimental facilities include in-core irradiation facilities (e.g., a glory hole), ex-core experiments, and associated auxiliary equipment.

10.4 DESIGN GUIDELINES

10.4.1 Guidelines for Determining the Minimum Safety Class of Experimental Facility Components

A. In-core experimental facilities (e.g., a glory hole in a burst reactor or a test loop in other pulsing reactors) should be designed to a safety class that is compatible with the safety requirements of the reactor core and (for other pulsing reactors only) the safety class of the reactor coolant system.

B. If the failure of an in-core fuel test loop (other pulsing reactors only) or other experimental facilities does not prevent safe shutdown (or cooling) of the reactor, but may result in potential off-site doses in excess of the 10 CFR 100 guidelines, the experimental facilities should be designed to a minimum safety class of SC-2.

Figure 10.1-1. Guide to the use of chapter 10.

C. If a failure of the experimental facility, or portions of it, do not prevent safe reactor shutdown, but may result in potential off-site doses in excess of 0.5 rem (but less than the 10 CFR 100 guidelines), it should be designed to a minimum safety class of SC-3.

D. Experimental facilities may be designed to non-nuclear standards if their failure does not prevent safe shutdown of the reactor or result in a potential off-site dose in excess of 0.5 rem.

10.4.2 Guidelines for the Design of Experimental Facilities

A. Experimental facility design should consider the potential reactivity effects of loading, unloading, movement, or failure of experimental equipment. Potential reactivity effects should be within the range for which the reactor core and the reactivity control system (see Sections 4.1 and 4.2) are designed.

B. The effects of experiments on flux measuring equipment should be considered when establishing protection system trip setpoints, and instrumentation and control system operating ranges (see Chapter 7).

C. Supports for experiments at burst facilities should be designed to prevent movement of the experiment during reactor operation (and, particularly, during the wait period prior to a burst).

D. The operation or failure of in-core experiments should not prevent safe reactor shutdown or cause physical damage to the reactor core assembly in excess of the guidelines in Section 4.1.

E. Experimental facilities should be designed to provide adequate protection against radiation from the following sources:

1. Fissionable experiments
2. Irradiated sample handling
3. Fuel failures within test loops (other pulsing reactors only)
4. Activated portions of experimental facilities
5. Other sources

Guidelines for occupational radiation exposures are included in Chapter 12.

F. The design of mechanical portions of experimental facilities should consider the guidelines in Section 3.9.

G. The instrumentation and controls associated with the experimental facility should include appropriate interlocks if operation of the experimental facility can create conditions that may adversely affect the safe operation of the reactor. If interlock circuits interface with the protection system, the guidelines in Section 7.1 should be considered in their design.

H. Strict review procedures for experiments and experiment plans should be established to ensure that proposed experiments satisfy the following general guidelines:

1. The proposed experiments should fall within limits for experiments delineated in the reactor facility technical specifications.

2. The proposed experiment should not present an unreviewed safety question.

3. The proposed experiment should not present an undue hazard to health and safety (see radiological consequence guidelines in Chapter 15).

4. The proposed experiment should not present an undue hazard to the reactor facility.

- I. Preburst administrative checks should be conducted, as necessary, to ensure that pulse or burst size will be no greater than the predetermined value set in by the reactivity control system.

10.4.3 Maintenance, Testing, and Inspection

- A. Prior to loading in the reactor core, experiments (particularly fissionable or explosive experiments) should be qualified for the expected environment by test, analysis, operating experience, or a combination of these methods. Testing of actual equipment under conditions which simulate the expected service conditions, to the extent practicable (with the exception of the radiation environment), is the preferred method.

- B. Interlock circuits between the reactor protection system or control systems and the experimental facilities should be designed to facilitate periodic testing to verify proper operation of the interlock.

- C. Actual experimental equipment should be closely scrutinized to ensure that it conforms with the intent of the experiment plan.

10.4.4 Interfaces With Other Systems

Experimental facilities may interface with the following other facility systems:

- A. Reactor Core Assembly

- B. Reactivity Control System

- C. Reactor Building

- D. Protection System

- E. Non-Safety I&C System

F. Non-Class IE Electric Power System

G. Radioactive Waste System

The design of experimental facilities should ensure a compatible application of codes and standards across all interfaces with other facility systems.

10.5 APPLICABLE GENERAL DESIGN CRITERIA

No General Design Criteria are specifically applicable to the design of experimental facilities. The GDCs that are applicable to interfacing systems should be considered in the design of experimental facilities that may potentially affect the safe operation of the reactor facility. In addition, the following provision of 10 CFR should be considered:

A. 10 CFR 50.59, "Changes, Tests and Experiments"

10.6 STANDARDS, CODES, AND GUIDES

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of experimental facilities:

A. Industry Standards and Codes

1. ANSI N16.1-1975, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors"

2. ANSI N394-1975 (ANS 14.1), "Operation of Fast Pulse Reactors"

3. ANSI N399-1974, "Records and Reports for Research Reactors"

4. ANSI N401-1974, "Review of Experiments for Research Reactors"

5. ANSI N405-1975, "Safety Guide for the Performance of Critical Experiments"

B. RDT Standards

The applicability of the requirements of the following RDT standard should be considered on a case basis:

1. E16-1T, "Test Vehicles for Transient Reactor Test Facility Experiments Containing Sodium"

C. Guides

1. USNRC Regulatory Guide 2.2, "Developments of Technical Specifications for Experiments in Research Reactors"

2. USNRC Regulatory Guide 2.4, "Review of Experiments for Research Reactors"

3. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors"

10.7 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. "Research Applications of Nuclear Pulsed Systems," International Atomic Energy Agency, Vienna, 1967

2. "Safe Operation of Critical Assemblies and Research Reactors," 1971 Edition, Safety Series No. 35, International Atomic Energy Agency, Vienna, 1971

3. "Manual for the Operation of Research Reactors," Technical Report, Series No. 37, International Atomic Energy Agency, Vienna, 1964

B. Pending Industry Standards and Codes

1. ANS 15.13, Proposed, "Allowed Core Alterations for Research Reactors"

2. ANS 15.18, Draft, "Standard for Administrative Controls for Research Reactors"

10.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 11

RADIOACTIVE WASTE SYSTEMS

11.1 SCOPE

The ERDA Manual, Chapter 0511, "Radioactive Waste Management," provides guidelines for establishing a program to manage radioactive waste in such a manner as to minimize the potential radiation exposure and associated risk to man. Chapter 0513 of the ERDA Manual, "Effluent and Environmental Monitoring and Reporting," prescribes monitoring requirements that are applicable to systems used to monitor radioactive waste areas. Guidelines for exposure to radiation and for maximum concentrations of radioactive material in air and water effluents are included in the ERDA Manual, Chapter 0524. These guidelines establish functional requirements for radioactive waste system designs. Radioactive waste material to be transported from a facility should be prepared for shipment in accordance with the provisions of the ERDA Manual, Chapter 0529, "Safety Standards for the Packaging of Fissile Material and Other Radioactive Material."

This chapter provides guidelines for implementing a radioactive waste management program in the design phase of reactor facility modifications or in the design of new reactor facilities. Figure 11.1-1 is a guide to the use of this chapter. DOE nonreactor nuclear facilities that handle radioactive waste are not subject to the recommendations in this chapter. These nonreactor nuclear facilities should consult ERDAM 0531 for pertinent guidance.

11.2 SAFETY FUNCTION

The safety function of the radioactive waste system is to collect and process radioactive waste material in such a manner that exposure to on-site and off-site personnel is maintained within the guidelines for the appropriate facility operating condition (see Chapter 15).

11.3 DEFINITION

The radioactive waste system includes equipment necessary to collect, store, sample, and treat radioactive waste materials and prepare them for reuse or disposal. Due to the characteristics of Category V reactors, the radioactive wastes are usually gaseous (e.g., fission product gases and airborne particulates released to the reactor building during reactor operation) or solid (e.g., fuel plates, contaminated tools). Liquid wastes are generated only in small quantities by experiments containing liquid or by decontamination residues. Solid and liquid wastes are handled by the radioactive waste system. Gaseous wastes are generally released to the atmosphere after filtering by the confinement exhaust filter banks described in Section 6.2.

Figure 11.1-1. Guide to the use of chapter 11.

The radioactive waste system also includes monitoring and control equipment necessary for ensuring that effluents released from the radioactive waste system will not result in doses or concentrations in excess of the guidelines for the applicable facility operating condition (see Chapter 15).

11.4 DESIGN GUIDELINES

11.4.1 Guidelines for Determining the Minimum Safety Class of Radioactive Waste System Components

Category V reactors do not generally require a safety class radioactive waste system. The off-site doses resulting from a failure of the radioactive waste system equipment used to process solid and liquid wastes will not generally exceed 0.5 rem.

11.4.2 Guidelines for the Design of a Radioactive Waste System

A. In determining the capabilities required for a radioactive waste system, the following sources of potentially radioactive material should be considered:

1. Reactor
2. Confinement ventilation system filters
3. Fuel in storage
4. Fissile experiments
5. Contaminated components or equipment used for maintenance, repair or testing
6. Cleaning solvents and rags
7. Reactor building sumps
8. Experimental facilities

B. The radioactive waste system should be designed such that exposures resulting from releases during normal operations are maintained as low as reasonably achievable below the ERDA Manual, Chapter 0524 guidelines (see Chapter 12).

C. Monitoring of radioactive waste system effluents should be provided as described in Chapter 12.

D. The capability for reducing the volume of radioactive waste should be provided where practical.

11.4.3 Maintenance, Testing, and Inspection

A. Monitoring and control equipment should include provisions for periodic testing, inspection, and calibration.

11.4.4 Interfaces With Other Systems

The radioactive waste system may interface with the following other facility systems:

- A. Reactor Building
- B. Non-Safety I&C System
- C. Non-Class IE Electric Power System
- D. Fuel-Handling and Storage System
- E. Experimental Facilities
- F. Radiation Protection System

The design of the radioactive waste system should ensure a compatible application of codes and standards across all interfaces with other facility systems.

11.5 APPLICABLE GENERAL DESIGN CRITERIA

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs and other provisions of 10 CFR be considered in the design of radioactive waste systems:

- A. GDC 60, "Control of Releases of Radioactive Material to the Environment"
- B. GDC 61, "Fuel Storage and Handling and Radioactivity Control"
- C. GDC 63, "Monitoring Fuel and Waste Storage"
- D. GDC 64, "Monitoring Radioactivity Releases"
- E. 10 CFR 20, "Standards for Protection Against Radiation"
- F. 10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Materials in Effluents - Nuclear Power Plants"
- G. 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Plants"

H. 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light Water-Cooled Nuclear Power Reactor Effluents")

11.6 STANDARDS, CODES, AND GUIDES

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of radioactive waste systems:

A. Industry Standards and Codes

1. ANSI N5.8-1976, "Definition of Radioactive Waste Categories"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents of Light Water-Cooled Power Plants"

2. USNRC Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light Water-Cooled Nuclear Power Reactors"

11.7 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems," NUREG 75/087, September 1975
2. WASH-1202, "Plan for the Management of AEC-Generated Radioactive Wastes"

B. Pending Industry Standards and Codes

1. ANSI N498 (ANS 40.12), Proposed, "General Criteria for Waste Management"

2. ANSI N525 (ANS 40.11), Proposed, "Definition of Radioactive Waste Categories"

3. ANSI N647 (ANS 15.12), Proposed, "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents"

11.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 12

RADIATION PROTECTION

12.1 SCOPE

Requirements for radiation protection are specified in the following DOE Manual chapters:

- A. 0502, "Notification, Investigation and Reporting of Occurrences"
- B. 0513, "Effluent and Environmental Monitoring and Reporting"
- C. 0524, "Standards for Radiation Protection"
- D. 0525, "Occupational Radiation Exposure Information"
- E. 0545, "Nuclear Accident Dosimetry Program"

This chapter provides additional design guidelines for radiation protection during normal facility operations, and includes references to detailed design standards that are applicable to radiation protection systems and procedures. Radiological consequence guidelines during postulated accidents are included in Chapter 15. Figure 12.1-1 is a guide to the use of this chapter.

12.2 SAFETY FUNCTION

The safety function of a radiation protection system is to minimize the exposure of man to radiation, considering all potential sources of radiation and radioactive material.

12.3 DEFINITION

The hardware portion of a radiation protection system includes the components and systems required for monitoring, surveying, sampling, and testing the radiation environment of a facility and its effluents. The system generally includes provisions for alarming abnormal conditions. The radiation protection system hardware also includes permanent and temporary shielding installed to minimize personnel exposure to direct and scattered radiation.

The administrative portion of the radiation protection system includes a system of records, policies, and operating procedures which facilitate maintaining occupational and off-site exposures to radiation and radioactive material As Low As Reasonably Achievable (ALARA).

Figure 12.1-1. Guide to the use of chapter 12.

12.4 DESIGN GUIDELINES

12.4.1 Guidelines for the Design of a Radiation Protection System

A. The radiation protection system, incorporating administrative procedures and protective hardware, should be designed to assure that the exposure of personnel to radiation is maintained within the following guidelines:

1. Occupational Radiation Exposure Guidelines

Exposure of personnel in controlled areas should be maintained ALARA in accordance with the ERDA Manual, Chapter 0524, Part V, and USNRC Regulatory Guide 8.8.

2. Off-Site Radiation Exposure Guidelines

Exposure of persons in unrestricted areas to routine releases of radioactive material should be maintained ALARA in accordance with the ERDA Manual, Chapter 0524, Part V, and 10 CFR 50, Appendix I.

These guidelines are applicable during all normal operating conditions, including steady-state and pulsing reactor operations.

B. The facility design should ensure adequate protection from direct and scattered radiation and airborne sources. Methods of protection to be considered in facility design include the following:

1. Permanent or temporary shielding.
2. Reactor building ventilation systems (see Section 6.2).
3. Area radiation zones and occupancy restrictions.
4. Fixed or portable monitoring and alarm systems.
5. Equipment design to minimize surface contamination.

C. The radiation protection system may be required to monitor the following:

1. Reactor building radiation and airborne activity levels.
2. Radioactive waste areas.
3. Fuel-handling and storage areas.
4. The facility environment, including gaseous effluent release paths.
5. Plant personnel (using personal dosimeters).
6. Contamination on personnel and surfaces.

D. Pulsed radiation rate-dependent response features of dosimetry equipment should be considered in determining the adequacy of the radiation protection system at a Category V reactor facility.

12.4.2 Maintenance, Testing, and Inspection

Monitoring equipment should include provisions for periodic testing, inspection, and calibration.

12.4.3 Interfaces With Other Systems

The radiation protection system may interface with the following other facility systems:

- A. Reactor Building
- B. Fuel Handling and Storage System
- C. Experimental Facilities
- D. Radioactive Waste Systems

The design of the radiation protection system should ensure a compatible application of codes and standards across the interfaces with other facility systems.

12.5 APPLICABLE GENERAL DESIGN CRITERIA

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs and other provisions of the 10 CFR be considered in the design of a radiation protection system:

- A. GDC 60, "Control of Releases of Radioactive Materials to the Environment"
- B. GDC 63, "Monitoring Fuel and Waste Storage"
- C. GDC 64, "Monitoring Radioactivity Releases"
- D. 10 CFR 19, "Notices, Instructions and Reports to Workers; Inspections"
- E. 10 CFR 20, "Standards for Protection Against Radiation"
- F. 10 CFR 30, "Rules of General Applicability to Licensing of Byproduct Material"
- G. 10 CFR 40, "Licensing of Source Material"
- H. 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably

Achievable' for Radioactive Material in Light Water-Cooled Nuclear Power Reactors Effluents"

I. 10 CFR 70, "Special Nuclear Material"

J. 10 CFR 100, "Reactor Site Criteria"

12.6 STANDARDS, CODES, AND GUIDES

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of a radiation protection system:

A. Industry Standards and Codes

1. Radiation Protection System Equipment

- a. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Material of Nuclear Facilities"
- b. ANSI N13.3-1969, "Dosimetry for Criticality Accidents"
- c. ANSI N13.4-1971, "Specification for Portable X- and Gamma-Radiation Survey Instruments"
- d. ANSI N13.5-1972, "Performance Specification for Direct Reading and Indirect Readings Pocket Dosimeters for X- and Gamma-Radiation"
- e. ANSI N13.7-1972, "Criteria for Film Badge Performance"
- f. ANSI N13.10-1974, "On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents"
- g. ANSI N16.2-1969, "Criticality Accident Alarm System"
- h. ANSI N101.6-1972, "Concrete Radiation Shields"
- i. ANSI N319-1976, "Personnel Neutron Dosimeters"
- j. ANSI N322-1977, "Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters"
- k. ANSI N512-1974, "Protective Coatings (Paints) for the Nuclear Industry"
- l. ANSI N648-1975, "Specification for Aggregates for Radiation Shielding Concrete"
- m. ANSI N649-1975, "Descriptive Nomenclature of Constituents of Aggregates for Radiation Shielding Concrete"

n. ANSI/ANS 6.4-1977 (N403), "Guidelines for the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants"

2. Radiation Protection System Administration

a. ANSI N2.1-1969, "Radiation Symbol"

b. ANSI N2.3-1967, "Immediate Evacuation Signal for Use in Industrial Installations Where Radiation Exposure May Occur"

c. ANSI N12.1-1971, "Fissile Material Symbol"

d. ANSI N13.2-1969, "Administrative Practices in Radiation Monitoring"

e. ANSI N13.6-1966, "Practice for Occupational Radiation Exposure Records Systems"

f. ANSI N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants"

g. ANSI N18.9-1972, "Program for Testing Biological Shielding in Nuclear Power Plants"

h. ANSI N105-1973, "Definition of Terms Relating to Dosimetry"

i. ANSI N510-1975, "Testing Nuclear Air Cleaning Systems"

j. ANSI Z88.1-1969, "Procedures for Respiratory Protection"

k. ANSI/ANS 6.1.1-1977 (N666), "Neutron and Gamma-Ray Flux-to-Dose Rate Factors"

l. ANSI/ANS 15.11-1977 (N628), "Radiological Control at Research Reactor Facilities"

B. RDT Standards

The applicability of the requirements of the following RDT standard should be considered on a case basis:

1. E9-1T, "HEPA Filters"

C. Guides

1. Radiation Protection System Equipment

a. USNRC Regulatory Guide 1.52, "Design, Testing and Maintenance for ESF Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants"

b. USNRC Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants"

c. USNRC Regulatory Guide 3.9, "Concrete Radiation Shields"

d. USNRC Regulatory Guide 4.5, "Measurements of Radionuclides in the Environment--Sampling and Analysis of Plutonium in Soil"

e. USNRC Regulatory Guide 4.6, "Measurements of Radionuclides in the Environment--Strontium-89 and Strontium-90 Analysis"

f. USNRC Regulatory Guide 8.3, "Film Badge Performance Criteria"

g. USNRC Regulatory Guide 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters"

h. USNRC Regulatory Guide 8.12, "Criticality Accident Alarm Systems"

i. USNRC Regulatory Guide 8.14, "Personnel Neutron Dosimeters"

2. Radiation Protection System Administration

a. USNRC Regulatory Guide 1.8, "Personnel Selection and Training"

b. USNRC Regulatory Guide 1.16, "Reporting of Operating Information--Appendix A, Technical Specifications"

c. USNRC Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Material in Liquid and Gaseous Effluents from Light Water-Cooled Nuclear Power Plants"

d. USNRC Regulatory Guide 1.109, "Calculation of Annual Dose to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I"

e. USNRC Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water-Cooled Reactors"

f. USNRC Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water-Cooled Power Reactors"

g. USNRC Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I"

h. USNRC Regulatory Guide 2.1, "Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors"

- i. USNRC Regulatory Guide 3.2, "Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles"
- j. USNRC Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants"
- k. USNRC Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants"
- l. USNRC Regulatory Guide 8.1, "Radiation Symbol"
- m. USNRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"
- n. USNRC Regulatory Guide 8.5, "Immediate Evacuation Signal"
- o. USNRC Regulatory Guide 8.7, "Occupational Radiation Exposure Records System"
- p. USNRC Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Reasonably Achievable"
- q. USNRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Reasonably Achievable"
- r. USNRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection"

12.7 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

- 1. USNRC Standard Review Plan 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems," NUREG 75/087, September 1975
- 2. USNRC Standard Review Plan 12.1, "Assuring that Occupational Radiation Exposures are ALARA," *ibid.*
- 3. USNRC Standard Review Plan 12.2, "Radiation Sources," *ibid.*
- 4. USNRC Standard Review Plan 12.3, "Radiation Protection Design Features," *ibid.*
- 5. USNRC Standard Review Plan 12.4, "Dose Assessment," *ibid.*

6. USNRC Standard Review Plan 12.5, "Health Physics Program," *ibid.*
7. Branch Technical Position ETSB 11-2, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Reactor Plants," (attached to SRP 11.3), *ibid.*
8. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962
9. Schaffer, N.M., ed., "Reactor Shielding for Nuclear Engineers," USAEC, 1973
10. "Respiratory Protective Services for Use in Atmospheres Containing Radioactive Materials," USBM-23, U.S. Bureau of Mines, 1973
11. UL-586, "High Efficiency Particulate, Air Filter Units," Underwriter's Laboratories, Inc.
12. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.
13. Burchsted, C.A., et al., "Nuclear Air Cleaning Handbook," ERDA 76-21, Oak Ridge National Laboratory, March 31, 1976
14. NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Material," October 1976
15. IAEA Safety Guide SG-05, Draft, "Operational Aspects of Radiological Protection," International Atomic Energy Agency, Vienna
16. "Safe Operation of Critical Assemblies and Research Reactors," 1971 Edition, IAEA Safety Series No. 35, International Atomic Energy Agency, Vienna, 1971
17. MIL-F-51068D, "Filter, Particulate, High Efficiency, Fire Resistant," 1974
18. MIL-STD-282, "Filter Units, Protective Clothing, Gas Mask Components and Related Products: Performance Test Methods," 1956
19. NCRP Report No. 8, "Control and Removal of Radioactive Contamination in Laboratories"
20. NCRP Report No. 10, "Radiological Monitoring and Methods"
21. NCRP Report No. 22, "Maximum Permissible Body-Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure"
22. NCRP Report No. 23, "Measurement of Neutron Flux and Spectra for Physical and Biological Applications"

23. NCRP Report No. 25, "Measurement of Absorbed Dose of Neutrons and Mixtures of Neutrons and Gamma Rays"

24. NCRP Report No. 28, "A Manual of Radioactivity Procedures"

25. NCRP Report No. 30, "Safe Handling of Radioactive Materials"

26. NCRP Report No. 38, "Protection Against Neutron Radiation"

27. NCRP Report No. 39, "Basic Radiation Protection"

B. Pending Industry Standards and Codes

1. ANSI N17.1, Proposed, "Location and Range of Measurement of Continuous Area and Process Radiation Monitoring System for Light Water Reactors"

2. ANSI N323, Draft, "Radiation Protection Instrumentation Test and Calibration"

3. ANSI N328, Draft, "Control of Radioactive Surface Contamination of Materials, Equipment and Facilities to be Released for Uncontrolled Use"

4. ANSI N336 (ANS 18.2.1), Draft, "Methods for Inferring Environmental Doses"

5. ANSI N346 (ANS 6.6), Draft, "Calculation and Measurement of Direct and Scattered Radiation from Nuclear Power Plants"

6. ANSI N404 (ANS 6.5), Proposed, "Glossary of Terms in Shielding and Dosimetry"

7. ANSI N410 (ANS 6.1.2), Proposed, "Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations"

8. ANSI N647 (ANS 15.12), Proposed, "Design Objectives for, and Monitoring of, Systems Controlling Research Reactor Effluents"

9. ANSI N651 (ANS 6.7), Proposed, "Radiation Zoning for Design of Nuclear Power Plants"

10. ANSI N702 (ANS 18.1.2), Draft, "Radioactive Materials in Effluents from Light Water-Cooled Nuclear Power Plants"

11. ANSI N703 (ANS 18.1.3), Draft, "Monitoring of Radioactive Materials in Effluents from Light Water-Cooled Nuclear Power Plants"

12. ANS 58.4, Draft, "Criteria for the Radiological Technical Specifications for Nuclear Power Stations"

12.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

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CHAPTER 13

CONDUCT OF OPERATIONS

13.1 SCOPE

The following ERDA Manual Chapters provide guidance for the conduct of facility operations:

- A. 0502, "Notification, Investigation, and Reporting of Occurrences"
- B. 0530, "Nuclear Criticality Safety"
- C. 0531, "Safety of Nonreactor Nuclear Facilities"
- D. 0540, "Safety of AEC-Owned Reactors"
- E. 0550, "Operational Safety Standards"
- F. 0601, "Emergency Preparedness Programs"
- G. Immediate Action Directives 8401-6 and 8401-7

This chapter provides a listing of other standards and guides that are recommended for consideration in the conduct of facility operations.

13.2 APPLICABLE GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR be considered in the conduct of facility operations:

- A. 10 CFR 19, "Notices, Instructions, and Reports to Workers; Inspections"
- B. 10 CFR 50.34, "Contents of Applications, Technical Information"
- C. 10 CFR 50.54, "Conditions of Licenses"
- D. 10 CFR 50.59, "Changes, Tests, and Experiments"
- E. 10 CFR 50, Appendix E, "Emergency Plans for Production and Utilization Facilities"
- F. 10 CFR 55, "Operators' License"
- G. 10 CFR 73, "Physical Protection of Plants and Materials"

13.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in the conduct of facility operations:

A. Industry Standards and Codes

1. ANSI N13.2-1969, "Administrative Practices in Radiation Monitoring"
2. ANSI N15.8-1974, "Nuclear Material Control Systems for Nuclear Power Plant"
3. ANSI N15.26-1976, "Physical Protection of Special Nuclear Material Within a Facility"
4. ANSI N16.1-1969, "Nuclear Criticality Safety Standard for Operations with Fissionable Materials Outside Reactors"
5. ANSI N18.17-1973, "Industrial Security for Nuclear Power Plants"
6. ANSI N394-1975 (ANS 14.1), "Operation of Fast Pulse Reactors"
7. ANSI N399-1974, "Records and Reports for Research Reactors"
8. ANSI N401-1974, "Review of Experiments for Research Reactors"
9. ANSI N546-1976, "Medical Certification and Monitoring for Personnel Requiring Operator Licenses for Nuclear Power Plants"
10. ANSI/ANS 15.4-1977 (N380), "Selection and Training of Personnel for Research Reactors"

B. RDT Standards

The applicability of the requirements of the following RDT standards should be considered on a case basis:

1. F1-3T, "Preparation of Unusual Occurrence Reports"
2. F3-2T, "Calibration of Program Requirements"
3. F4-20T, "Operation and Maintenance Manuals"

C. Guides

1. USNRC Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage"
2. USNRC Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants"

3. USNRC Regulatory Guide 1.114, "Guidance for Being Operator at the Controls of a Nuclear Power Plant"
4. USNRC Regulatory Guide 1.134, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses"
5. USNRC Regulatory Guide 2.4, "Review of Experiments for Research Reactors"
6. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors"
7. USNRC Regulatory Guide 5.7, "Control of Personnel Access to Protected Areas, Vital Areas and Material Access Areas"
8. USNRC Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials"
9. USNRC Regulatory Guide 5.29, "Nuclear Material Control System for Nuclear Power Plants"
10. USNRC Regulatory Guide 5.51, "Management Review of Nuclear Material Control and Accounting System"
11. USNRC Regulatory Guide 5.54, "Standard Format and Content of Safeguards Contingency Plan for Nuclear Power Plants"
12. USNRC Regulatory Guide 5.57, "Shipping and Receiving Control of Special Nuclear Material"
13. USNRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"
14. USNRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Reasonably Achievable"

13.4 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. "Safe Operation of Critical Assemblies and Research Reactors, 1971 Edition," IAEA Safety Series No.35, International Atomic Energy Agency, Vienna, 1971

2. "Manual for the Operation of Research Reactors," IAEA Technical Report Series No. 37, International Atomic Energy Agency, Vienna, 1964

3. "Planning for the Handling of Radiation Accidents," IAEA Safety Series No. 32, International Atomic Energy Agency, Vienna, 1969

4. IAEA Safety Guide SG-01, Draft, "Staffing, Recruitment, Training, and Authorization of Operating Personnel," International Atomic Energy Agency, Vienna

5. USNRC Standard Review Plan 13.1.2, "Operating Organization," NUREG 75/087, September 1975

6. USNRC Standard Review Plan 13.1.3, "Qualifications of Nuclear Plant Personnel," *ibid.*

7. USNRC Standard Review Plan 13.2, "Training," *ibid.*

8. USNRC Standard Review Plan 13.3, "Emergency Planning," *ibid.*

9. USNRC Standard Review Plan 13.4, "Review and Audit," *ibid.*

10. USNRC Standard Review Plan 13.5, "Plant Procedures," *ibid.*

11. USNRC Standard Review Plan 13.6, "Industrial Security," *ibid.*

12. EPA 520/1-75-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," U.S. Environment Protection Agency, September 1975

13. NUREG-0094, "A Guide to the Licensing of Facility Operators, Including Senior Operators," USNRC, July 1976

14. WASH 1130, Revised, "Utility Staffing and Training for Nuclear Power," USAEC, June 1973

B. Pending Industry Standards and Codes

1. ANSI N17.2 (ANS 15.16), Draft, "Standard for Emergency Planning for Research Reactors"

2. ANSI N682 (ANS 3.7.1), Proposed, "Facilities for Medical Care On and Off Plant Site"

3. ANSI N684 (ANS 3.7.2), Proposed, "Emergency Control Centers On and Off Plant Site"

4. ANSI N685 (ANS 3.7.3), Proposed, "Guidance on Conducting Emergency Preparedness Drills"

5. ANSI N700 (ANS 15.14), Draft, "Standard for Physical Security of Research Reactors"

6. ANS 15.18, Draft, "Standard for Administrative Controls for Research Reactors"

13.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

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CHAPTER 14

INITIAL TEST PROGRAM

14.1 SCOPE

10 CFR 50, Appendix B, Section XI establishes the requirements for an initial startup test program to verify that all structures, systems, and components will satisfactorily perform their safety functions at a nuclear facility. The responsibilities and authorities for assuring the adequate protection of health and safety during the initial reactor operations are included in the ERDA Manual, Chapter 0540, "Safety of AEC-Owned Reactors."

This chapter provides a listing of other standards and guides that are recommended for consideration in the conduct of the initial test program.

14.2 APPLICABLE GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR be considered in the conduct of the initial test program:

- A. 10 CFR 50.34, "Contents of Applications: Technical Information"
- B. 10 CFR 50, Appendix B, Section XI, "Test Control"

14.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following codes, standards, and guides be considered in the conduct of the initial test program:

- A. Industry Standards and Codes
None
- B. RDT Standards
None
- C. Guides
None

14.4 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and

standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. IAEA Safety Guide SG04, Draft, "Commissioning Procedures,"
International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

1. ANSI N267 (ANS 3.6), Proposed, "Requirements for Preoperational and Startup Testing"

2. ANS 15.18, Draft, "Standard for Administrative Controls for Research Reactors"

14.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 15

ACCIDENT ANALYSIS

15.1 SCOPE

The ERDA Manual, Chapter 0540, "Safety of AEC-Owned Reactors," requires that the safety of each reactor facility be analyzed, evaluated and documented. This chapter provides general guidelines for performing accident analysis and for applying the single-failure criterion. This chapter also recommends standards and guides to be considered in the analysis of postulated accidents at a reactor facility.

15.2 GUIDELINES FOR ACCIDENT ANALYSIS

A. A reactor facility should be designed to accommodate postulated accidents (the initiating event from Table 15.2-1 that starts the accident sequence plus any other events or failures that are a direct or consequential result of the initiating event) within the Table 15.2-2 consequence guidelines for the operating condition that corresponds to the initiating event. With the addition of a single failure (discussed in Section 15.3), the reactor facility should generally accommodate the postulated accident plus single failure within the consequence guidelines for the next higher operating condition (e.g., an emergency event plus a single failure may be accommodated within the faulted consequence guidelines). A faulted event plus a single failure should, however, still be accommodated within the consequence guidelines for the faulted operating condition. If analysis indicates that the facility cannot meet the appropriate consequences guidelines of Table 15.2-2 for all design basis accidents, some system design changes may be warranted.

B. The radiological consequences of the postulated accidents should not exceed the guidelines in Table 15.2-2. The dose guidelines for the faulted facility operating condition are discussed in detail below.

1. On-Site Dose Guidelines

In determining the on-site dose consequences that may result from postulated design basis accidents, appropriate credit may be taken for the site emergency plan and the ability to transport and evacuate on-site personnel. Considering this ability to manage on-site personnel, it is recommended that the accident dose consequence goals be considerably less than the faulted operating condition dose guidelines in Table 15.2-2 (25 rem total radiation dose to the whole body, 300 rem total radiation dose to the thyroid from iodine exposure, 150 rem to the bone and 75 rem to the lung from plutonium exposure, or 75 rem to any other organ).

For individuals who constitute the site emergency force, the guidance in ERDAM 0524 (particularly in Part IV, "Guidance for Emergency Exposures During

Table 15.2-1

Spectrum of Postulated Accidents to be Considered
in the Design of Category V Reactor Facilities

A. REACTIVITY ACCIDENTS

1. Inadvertent Criticality During Core Assembly or Maintenance (Burst Reactors Only)
2. Inadvertent Criticality During Startup
3. Reactivity Control System Misadjustment
4. Spectrum of Reactivity Control System Failures
 - a. Unscheduled pulse or burst
 - b. Pulse or burst larger than planned
5. Movement of Experiments
 - a. At power
 - b. During wait period (Burst Reactors Only)

B. OTHER ACCIDENTS

1. Loss of Electric Power
2. Experiment Failures
 - a. Failure of fissionable experiment
 - b. Detonation of explosive experiment
 - uncontained explosive
 - fully contained explosive
 - c. Fission foil vaporization
 - d. Other experiment failures
3. Test Loop Failures (Other Pulsing Reactors Only)
4. Loss of Reactor Building Ventilation System
5. Loss of Core Cooling System, e.g., an N₂ inerting/cooling system, if such a system is provided. (Burst Reactors Only)
6. Fire at Reactor Facility
 - a. In reactor building
 - b. In fuel storage vault
 - c. Other fires

Table 15.2-2

Summary of Facility Operating Conditions
and the Corresponding Frequency of
Occurrence and Consequence Guidelines

Operating Condition	Description	Typical Frequency of Occurrence, F (yr ⁻¹)*	Typical Design Basis Events	Radiological Consequence Guidelines	
				Off-Site	On-Site
Normal	Events that are planned to occur regularly in the course of plant operation	$F \geq 1$	Startup, shutdown, normal reactivity pulses	ALARA** ($\leq$ ERDAM 0524)	ALARA** $\leq$ ERDAM 0524
Upset	Incidents or occurrences of moderate frequency which may occur several times during the life of a facility	$1 > F \geq 10^{-1}$	Loss of electric power, inadvertent reactor shutdown	ALARA** ($\leq$ ERDAM 0524)	ALARA** $\leq$ ERDAM 0524
Emergency	Infrequent incidents or occurrences which may occur during the lifetime of a facility	$10^{-1} > F \geq 10^{-3}$	Reactivity burst larger than planned	Small fraction of 10CFR100	Small fraction of on-site faulted dose guidelines
Faulted	Events which are limiting faults and are not expected to occur during the life of a facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material and because they represent upper bounds on failures or accidents with a probability of occurrence sufficiently high to require consideration in design. Generally a postulated accident is not considered in design if the probability of its occurrence resulting in unacceptable consequences is less than 10^{-7} per year	$10^{-3} > F \geq 10^{-7}$	Design basis reactivity accident	$\leq 10CFR100$	≤ 25 rem whole body, ≤ 300 rem thyroid, ≤ 150 rem bone, ≤ 75 rem lung, and ≤ 75 rem to other organs

* The tabulated frequency ranges for each operating condition are illustrative examples only. They are based on the written description in the table, for a facility with an expected lifetime in the 20-40 year range.

** As low as reasonably achievable.

Table 15.2-2 (Continued)

Summary of Facility Operating Conditions
and the Corresponding Frequency of
Occurrence and Consequence Guidelines

Operating Condition	General Guidelines	Fuel Design Guidelines	Mechanical System Design Guidelines
Normal	No protection system action required	No fuel damage (see Section 4.1)	Maximum equipment loading within applicable design code allowables (see Section 3.9)
Upset	The facility should be capable of returning to operation without extensive correction action or repair		
Emergency	Facility should be capable of returning to operation following corrective action or repair of damage	Any fuel damage should not prevent reactor shutdown (see Section 4.1)	
Faulted	Facility damage may preclude return to operation		

15-1(c)

Rescue and Recovery Activities,") should be considered in determining the dose guidelines that are applicable to emergency force members when performing their duties during an emergency in response to a postulated design basis accident.

2. Off-Site Dose Guidelines

a. At the exclusion area boundary (see Section 2.1), an individual remaining at any point on the boundary for two hours immediately following the onset of the design basis accident will not receive a total whole body radiation dose in excess of 25 rem, a total thyroid dose in excess of 300 rem from iodine exposure, 150 rem to the bone and 75 rem to the lung from plutonium exposure, or 75 rem to any other organ.

b. At the outer boundary of the low population zone (see Section 2.1), an individual located at any point on the outer boundary exposed to the radioactive cloud from the postulated fission product release resulting from the design basis accident (during the entire period of its passage) will not receive a total whole body radiation dose in excess of 25 rem, a total thyroid dose in excess of 300 rem from iodine exposure, 150 rem to the bone and 75 rem to the lung from plutonium exposure, or 75 rem to any other organs.

3. Dose Equivalence

Following some postulated accidents, the dose to some critical organ due to inhalation or ingestion or radionuclides may be more limiting than the whole body or thyroid dose guidelines for the appropriate operating condition. For these accidents, the critical organ should be identified and the organ dose should be limited such that it does not exceed a value that would be comparable to the corresponding whole body dose or thyroid dose guidelines.

C. The minimum safety class to which components, systems, and structures should be designed is determined according to the guidelines in Section 3.2. As recommended in that section, a system which contains, or may contain, radioactivity may be designed to non-nuclear standards if the maximum off-site radiological consequences of system failure do not exceed 0.5 rem to the whole body and if the system is not otherwise required to be Safety Class 1, 2 or 3 (see Table 3.2-1). Likewise, a system should be designed to a minimum of Safety Class 3 if the maximum off-site radiological consequences of system failure exceed 0.5 rem to the whole body and the system is not otherwise required to be Safety Class 1 or 2. As described in Section 3.2, Category V burst reactors do not generally require components, systems or structures to be designed to Safety Class 1 or 2 standards. Other Category V pulsing reactors (e.g., tank-type reactors) may have SC-1 or -2 components associated with the reactor facility or with experimental test loops (see appropriate other Design Guide).

The final safety classification of a system is obtained by analysis of the design basis accidents. Because of specific features of an individual Category V reactor facility, analysis of the design basis accidents may indicate that some consequences are not within the guidelines of Table 15.2-2. Therefore, some system design changes may be necessary (i.e., improved reactor core design) to ensure that a final safety class equivalent to the recommended

minimum safety class will result in consequences within the guidelines of Table 15.2-2. Alternatively, it may be necessary to increase the quality and reliability of the system by designing to a final safety class that is higher than the minimum safety class recommended by this Design Guide.

D. The spectrum of design basis accidents to be considered in the design of Category V reactors is listed in Table 15.2-1. Events such as floods, storms, and earthquakes are not explicitly considered in this chapter. Severe natural phenomena (floods, storms, earthquakes) are discussed in Chapter 3. Probability guidelines applicable to the determination of the spectrum of severe natural phenomena to be considered in reactor facility design are described in Sections 3.1 and 3.2.

On the basis of the likelihood of occurrence during the life of a reactor facility, design basis accidents may be classed as upset, emergency, or faulted operating conditions.

The descriptions of the facility operating conditions in Table 15.2-2 include very broad guidelines for determining the facility operating condition to which a particular accident should be assigned. On the basis of these broad guidelines, Table 15.2-2 additionally provides numerical examples of the range of annual frequency of occurrence that may typically be expected for accidents in each of the facility operating conditions. The actual frequency of occurrence range corresponding to each operating condition should be determined for each individual reactor facility on the basis of the expected facility operating lifetime and the nature and consequences of accidents postulated for the facility.

This Design Guide identifies General Design Criteria from 10 CFR 50, Appendix A, that may be applicable to Category V reactor facility design. These criteria refer to Anticipated Operational Occurrences (AOOs). Appendix A of 10 CFR 50 defines AOOs as events which are expected to occur one or more times in the life of the reactor facility. Therefore, AOOs include normal, upset, and the more likely emergency events. The relationship between AOOs and these facility operating conditions is illustrated in Figure 15.2-1.

E. The spectrum of design basis accidents and severe natural phenomena considered in facility design is acceptable if it includes all postulated types of accidents for which a realistic estimate of the probability of occurrence of potential doses comparable to the faulted operating condition guidelines exceeds approximately 10^{-7} per year.

In view of the low probability events under consideration, the probability of occurrence of the initiating events leading to these potential radiological consequences should be estimated using assumptions that are as realistic as is practicable. In addition, because of these low probability events, valid statistical data are often not available to permit an accurate quantitative calculation of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of potential doses comparable to the faulted operating condition guidelines is approximately 10^{-6} per year is acceptable, if, when

Note: These frequency ranges for each operating condition and for AOOs are illustrative examples only. They are based on the information in Table 15.2-2.

Figure 15.2-1. Approximate relationship between the facility operating conditions and the anticipated operational occurrences.

combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

The overall safety objective for an individual facility should be to ensure that there is less than one chance in a million per year ($< 10^{-6}$ per year) of a serious accident.

F. The accident analyses should make conservative assumptions regarding the values of all parameters which may have an effect on the calculated consequences of postulated accidents. Some particular conservatisms to be included in the accident analysis are listed below:

1. The short term diffusion estimates used for calculating the radiological consequences of postulated accidents should consider the guidance in USNRC Standard Review Plan 2.3.4 regarding the acceptability of models and conservatisms appropriate to the design basis calculations.

2. The initial conditions for the accident analysis (before the initiating event) should be within the range of operating conditions permitted by the facility technical specifications.

3. Conservative time intervals for safety-related operator actions that are required to mitigate the consequences of postulated accidents should consider the guidance in ANSI N660.

15.3 GUIDELINES FOR APPLICATION OF THE SINGLE FAILURE CRITERION TO ACCIDENT ANALYSIS AND SYSTEM DESIGN

A. General

As defined in 10 CFR 50, Appendix A, a single failure is an occurrence resulting in the loss of capability of a component to perform its intended safety function. Multiple failures resulting from a single occurrence are considered a single failure. To ensure a conservative approach to the design of safety class systems and to the analysis of the consequences of postulated accidents of water-cooled power reactors, the General Design Criteria require that systems such as the protection system, electric power system, and other safety class systems be designed such that the system safety function can be accomplished with a single failure in the safety class system. This is the basic single failure criterion. Its application to system design assists in determining the minimum system redundancy required to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. An objective in the design of a safety class system is to ensure that a single failure in a component or channel will not interfere with the proper operation of its redundant counterpart.

As described in previous chapters, Category V reactors have very different system requirements than commercial power reactors. Category V reactors have few safety class systems and those which are provided are functionally simpler than those in power reactors. This is due largely to the inherent safety

features incorporated in Category V reactor design. The safety systems that are provided, however, should still be designed such that the system safety function can be accomplished with a single failure in the safety class system.

B. Types of Single Failures

Single failures may occur in either active or passive components. An active component is defined as one which requires physical motion of some sort to perform its intended function (e.g., motor operated valves or electrical relays). Failure of an active component means failure to provide the physical motion required for the safety function, not the failure of the passive parts (e.g., a valve body) of the active component. A passive component, such as a pipe, tank, accumulator, or electrical cable, does not require physical motion to perform its function.

Active single failures may include failure to act, or incorrect action when required, and spurious action or operation when not required. Examples of single active failures are failure of a valve to open or close in response to a valid signal, and the operation of a motor-operated valve in the absence of a valid signal. Passive single failures may include ruptures or other failures of fluid system pressure boundaries (e.g., a pneumatic line), flow blockage in fluid systems, and electric short or open circuits.

C. Application of the Single Failure Criterion

Reactor facilities are designed to accommodate a spectrum of postulated accidents without causing undue risk to health and safety. The basic procedure for conservatively determining the effects of a postulated accident and the effectiveness of safety class systems in mitigating potential consequences include the following steps:

1. The initiating event is specified. It may be assumed to occur in a safety class system (e.g., a reactivity transient caused by an uncontrolled positive reactivity insertion due to a fault in the reactivity control system).
2. Other failures that are a direct consequence of the initiating event are then assumed to take place.
3. In response to the postulated accident, safety class systems may be called into operation to mitigate the accident consequences. To ensure a conservative design of these systems, a single failure is assumed to occur in a safety class system, thereby diminishing its capabilities. Because of the inherent safety features of Category V reactors, few safety class systems are required.
4. Consequences of the postulated accident must then be demonstrated to be within the corresponding guidelines specified in Table 15.2-2.

10 CFR 50, Appendix A, states that fluid and electric systems are considered to be designed against an assumed single failure if neither a single failure of any active component (assuming passive components function properly) nor a single failure of a passive component (assuming active components function

properly) results in a loss of the capability of the system to perform its safety function.

Some single failures, such as a failure of a standby electric power supply, may disable a protection system channel and any redundant safety system trains powered from the power supply. This multiple failure, resulting from a single occurrence, is still considered to be a "single failure."

The basic statement of the single-failure criterion is concise; however, its effects are far reaching and often require interpretation to permit reasonable implementation in the design of reactor facility systems. Additional guidance on the application of the single-failure criterion is found in USNRC Regulatory Guide 1.53 and IEEE 379.

Safety class system failures may be classified as either detectable or undetectable failures. All potential single failures are detectable failures and may be discovered by periodic tests, anomalous indications, or alarms. For the purpose of system analysis, all potential undetectable failures should be identified by means of Failure Mode and Effects Analysis (FMEA). Undetectable failures differ from detectable failures in that they may exist in a safety class system for years. In fact, they could be built into the system by the manufacturer or installer. When undetectable failures are identified, the preferred course of action is to redesign the safety class system or the testing program to eliminate potential undetectable failures. An alternative approach is to perform the analysis of the effects of each single failure while assuming that all undetectable failures are in the failed mode. If this analysis indicates that the safety class system can still complete its required safety function and satisfactorily mitigate the consequences of the postulated design basis accident, then certain undetectable failures may be an acceptable feature of the system design.

For the purpose of applying the single-failure criterion, the period following a postulated accident is generally divided into two phases - the "short term" and the "long term." During the short term, the facility should be designed to accommodate the postulated accident within the applicable consequence guidelines with the additional occurrence of (1) any electrical single failure, or (2) any active fluid system failure (e.g., failure of a valve to close).

Due to the relatively brief nature of Category V reactor accidents, there is no "long term" post-accident period during which safety class systems must remain operable. Therefore, application of the single-failure criterion during the "long term" is not applicable to Category V reactors.

15.4 APPLICABLE GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR be considered when performing or evaluating the analysis of postulated accidents at a reactor facility:

A. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

B. 10 CFR 100, "Reactor Site Criteria"

15.5 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered when performing or evaluating the analysis of postulated accidents at a reactor facility:

A. Industry Standards and Codes

1. ANSI N660 (ANS 51.4), Draft, "Criteria for Safety-Related Operator Actions"

2. IEEE 379-1977 (ANSI N41.2), "Application of the Single-Failure Criterion to Nuclear Power Generating Station Class IE Systems"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems"

2. USNRC Regulatory Guide 1.105, "Instrument Spans and Setpoints"

15.6 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. Pitkanen, P.H., "Hazards of Fast Burst Facilities," Nuclear Safety, Vol. 4, No. 1, September 1962

2. USNRC Standard Review Plan 2.2.3, "Evaluation of Potential Accidents," NUREG 75/087, September 1975

3. USNRC Standard Review Plan 2.3.4, "Short Term Diffusion Estimates," *ibid.*

4. Branch Technical Position EICSB 18, "Application of the Single Failure Criteria to Manually Controlled Electrically Operated Valves," (in SRP, Appendix 7A), *ibid.*

5. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962

6. Slade, D.H., (Editor), "Meteorology and Atomic Energy - 1968," TID-24190, USAEC, 1968

7. Briggs, G.A., "Plume Rise," TID-25075, USAEC, 1969

8. Turner, D.B., "Workbook on Atmospheric Dispersion Estimates," AP-26, Revised, USEPA Office of Air Programs, 1970

9. Streng, D.L., et al., "Review of Calculational Models and Computer Codes for Environmental Dose Assessment of Radioactive Releases," BNWL-B-454, Battelle Northwest Laboratories, June 1976

10. Winton, M.L., "Computer Codes for Analyzing Nuclear Accidents," Nuclear Safety, Vol. 15, No. 5, September - October 1974

B. Pending Industry Standards and Codes

1. ANS 2.16, Draft, "Guidelines for Analysis of Atmospheric Transport of Accidental Release of Nuclear Reactor Effluents"

15.7 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

The following topical listing identifies areas in which the development of additional nuclear standards may provide a more consistent approach to Category V reactor facility design and modification:

A. Guidelines for a standard approach to calculating doses from postulated DOE reactor accidents. (This guide should define the minimum level of conservatism to be incorporated into accident analyses.)

CHAPTER 16

TECHNICAL SPECIFICATIONS

16.1 SCOPE

The ERDA Manual, Chapter 0540, "Safety of AEC-Owned Reactors," requires that the limitations for safe operation be specified in a Technical Specification document for each reactor, with the detailed requirements being commensurate with the potential risks that may arise from facility operation.

This chapter provides a listing of other guides and standards that are recommended for consideration when establishing technical specification limits for safe reactor operation.

16.2 APPLICABLE GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR be considered in establishing technical specifications for reactor facilities:

A. 10 CFR 50.36, "Technical Specifications"

B. 10 CFR 50, Appendix I, Section IV, "Guides on Technical Specifications for Limiting Conditions for Operation for Light Water-Cooled Power Reactors Licensed Under 10 CFR Part 50"

16.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in establishing technical specifications for reactor facilities:

A. Industry Standards and Codes

1. ANSI N378-1974, "Standards for Development of Technical Specifications for Research Reactors"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.16, "Reporting of Operating Information," Appendix A, Technical Specifications"

2. USNRC Regulatory Guide 1.105, "Instrument Setpoints"

3. USNRC Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors"

4. USNRC Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants"

16.4 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. "Guide to Content of Technical Specifications for Nuclear Reactors," USAEC, November 1968

2. USNRC Standard Review Plan 2.4.14, "Technical Specifications and Emergency Operation Requirements (Hydrology)," NUREG 75/087, September 1975

3. IAEA Safety Guide SG-03, Draft, "Safety Guide on Operational Limits and Conditions," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

1. ANS 58.4, Draft, "Criteria for the Radiological Technical Specifications for Nuclear Power Stations"

16.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 17

QUALITY ASSURANCE

17.1 SCOPE

The ERDA Manual, Chapter 0540, "Safety of AEC-Owned Reactors," requires that quality assurance programs be developed and implemented for reactor projects. The implementation of the Quality Assurance program is described in the ERDA Manual, Chapter 0820, "Quality Assurance."

This chapter lists other standards and guides that are recommended for consideration when establishing QA programs for safety class components, systems, and structures during the design, constructions, and operation phases of a reactor facility. Non-safety class (non-nuclear) components, systems, and structures may utilize normal industrial quality assurance procedures in lieu of the codes, standards, and guides listed in this section, providing they meet the requirements of ERDA 0820.

17.2 APPLICABLE 10 CFR GENERAL CRITERIA

It is recommended that the following General Design Criterion and other provisions of 10 CFR 50 be considered in establishing quality assurance programs for the design, construction, and operation phases of reactor facilities:

A. GDC 1, Quality Standards and Records"

B. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

17.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in establishing a quality assurance program:

A. Industry Standards and Codes

1. General QA (applicable to all phases)

a. ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities"

b. ANSI N45.2.9-1974, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants"

c. ANSI N45.2.10-1973, "Quality Assurance Terms and Definitions"

Reactors" d. ANSI N402-1976, "Quality Assurance Program for Research

e. ANSI/ASTM E380-76/IEEE 268-1976, "Metric Practice"

2. Design Phase QA

a. ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

3. Construction Phase QA

a. ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"

4. Operation Phase QA

a. ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants"

b. ANSI N399-1974, "Records and Reports for Research Reactors"

B. RDT Standards

The applicability of the requirements of the following RDT standards should be considered on a case basis:

1. General QA (applicable to all phases)

a. F1-2T, "Preparation of System Design Descriptions"

b. F2-2, "Quality Assurance Program Requirements"

2. Design Phase QA

a. F2-9T, "Reliability Assurance"

3. Construction and Operation Phase QA

a. F2-4T, "Quality Verification Program Requirements"

b. F2-7T, "Acceptance Sampling Plans"

c. F3-2T, "Calibration Program Requirements"

C. Guides

1. General QA (applicable to all phases)

a. USNRC Regulatory Guide 1.74, "Quality Assurance Terms and Definitions"

b. USNRC Regulatory Guide 1.88, "Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records"

c. USNRC Regulatory Guide 2.5, "Quality Assurance Program Requirements for Research Reactors"

2. Design Phase QA

a. USNRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)"

b. USNRC Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

3. Construction Phase QA

a. USNRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)"

4. Operation Phase QA

a. USNRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)"

b. USNRC Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment"

17.4 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 17.1, "Quality Assurance During the Design and Construction Phase," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 17.2, "Quality Assurance During the Operations Phase," *ibid.*

3. WASH 1283, Revision 1, "Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," (Grey Book), USAEC, May 1974

4. WASH 1284, "Guidance on Quality Assurance Requirements During the Operational Phase of Nuclear Power Plants," (Orange Book), USAEC, May 1974

5. WASH 1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," (Green Book), USAEC, October 1973

6. "Matrix of Nuclear quality Assurance Program Requirements," American Society for Quality Control, April 1973

7. "Quality Systems in the Nuclear Industry," STP 616, American Society for Testing and Materials

8. IAEA Safety Guide SG-QA1, Draft, "Quality Assurance Program Preparation," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

1. ANSI N45.2.12, Draft, "Requirements for Auditing of QA Programs for Nuclear Power Plants"

2. ANSI N45.2.15, Draft, "Requirements For the Control of Hoisting, Rigging, and Transporting of Items at Nuclear Power Plant Site"

17.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 18

Distribution List

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