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DESIGN GUIDE

FOR

CATEGORY VI REACTORS

AIR COOLED GRAPHITE REACTORS

MASTER

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

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DESIGN GUIDE FOR CATEGORY VI REACTORS AIR-COOLED GRAPHITE REACTORS

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February 1979

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CONTENTS

CHAPTER		PAGE
	FOREWORD	vi
1	INTRODUCTION	1.1
2	SITE CHARACTERISTICS	
	2.1 Geography and Demography	2.1
	2.2 Meteorology	2.3
	2.3 Hydrology	2.6
	2.4 Geology and Seismology	2.9
3	DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS	
	3.1 Introduction to Design Classification	3.1
	3.2 Classification of Structures, Components and Systems	3.3
	3.3 Wind and Tornado Loadings	3.8
	3.4 Water Level (Flood) Design	3.10
	3.5 Missile Protection	3.12
	3.6 Protection Against the Dynamic Effects Associated with Postulated Rupture of Piping	3.15
	3.7 Seismic Design	3.15
	3.8 Design of Category I Structures	3.18
	3.9 Mechanical Systems and Components	3.22
	3.10 Seismic Qualification of Category I Instrumentation and Electric Equipment	3.24
	3.11 Environmental Design of Mechanical and Electrical Equipment	3.26
	3.12 Welding of Nuclear Components	3.29
	3.13 Nuclear Reactor Facility Materials	3.30

CHAPTER	PAGE
3.14 Decommissioning of Reactor Facilities	3.31
4 REACTOR	
4.1 Reactor Core Assembly	4.1
4.2 Reactivity Control System	4.4
5 REACTOR COOLING SYSTEM	5.1
6 ENGINEERED SAFETY FEATURES	
6.1 Reactor Building	6.1
6.2 Engineered Safety Feature Ventilation System	6.8
7 INSTRUMENTATION AND CONTROLS	
7.1 Protection System and Safety Class Instrumentation	7.1
7.2 Non-Safety Class Instrumentation and Control Systems	7.9
8 ELECTRIC POWER	
8.1 Class IE Electric Power Systems	8.1
8.2 Non-Class IE Electric Power Systems	8.1
9 AUXILIARY SYSTEMS	
9.1 Fuel Handling and Storage Systems	9.1
9.2 Fire Protection Systems	9.5
10 EXPERIMENTAL FACILITIES	10.1
11 RADIOACTIVE WASTE SYSTEMS	11.1
12 RADIATION PROTECTION	12.1
13 CONDUCT OF OPERATIONS	13.1
14 INITIAL TEST PROGRAM	14.1
15 ACCIDENT ANALYSIS	15.1
16 TECHNICAL SPECIFICATIONS	16.1

CHAPTER	PAGE
17	QUALITY ASSURANCE 17.1
18	DISTRIBUTION LIST AND COMMENT SHEETS 18.1

FOREWORD

The Department of Energy (DOE) owns and operates a wide variety of nuclear reactors, including air-cooled graphite reactor 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 in a manner that gives adequate consideration 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 the 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 operating power levels, lower fission product inventories, and reduced dependence on the operation of safety systems required to assure safe reactor shutdown and cooling and to assure acceptable consequences of postulated accidents. As a result, simplified and/or unique 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, siting, operating characteristics, and 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 is a list of the DOE reactors considered in writing this guide for air-cooled graphite reactors.

For each reactor category, a Design Guide tailored to provide design and safety guidelines for systems typical of reactors within the particular category is being developed to assist the facility contractor. 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, guidelines are provided for determining if the system is required at a particular facility, and when the system is required, the Design Guide further describes the major safety concepts and design philosophies to be implemented in the minimum design of 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

Table A-1

DOE 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 Reactors Considered in the
Category VI Design Guide

REACTOR	MAXIMUM STEADY STATE THERMAL POWER	NOTES
TREAT	100 KW	Transient and Burst Features ⁽¹⁾
SR-305	50 W	

⁽¹⁾ The transient and burst features are addressed in the Design Guide for Category V, Burst and Pulse Type Reactors.

specifications, and quality assurance. For each reactor system, and for each of the general topical areas described above, the Design Guide provides a comprehensive listing of the applicable portions of 10 CFR (including the General Design Criteria in 10 CFR 50, Appendix A), and the industry and NRC standards, codes, and guides 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 applicable to the design of air-cooled graphite reactor facilities. The Design Guide includes detailed design guidelines which provide interpretations of the standards, codes, and guides in order to facilitate their application to air-cooled graphite reactor facilities. When criteria from existing standards, codes, and guides do not provide adequate design guidance, the guidelines presented in this Design Guide serve as interim criteria pending the development or approval of appropriate new standards, codes, and guides.

Aided by the introductory guidance provided by 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.

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. In this manner, the Design Guide will remain a timely standard reference for the designer and reviewer alike.

To facilitate feedback from the users of this guide, Chapter 18 includes preaddressed comment sheets. The purpose of these sheets is to provide a convenient method for the user to note and transmit comments or suggestions that he may generate in using the guide. These comments and suggestions will then be considered for appropriate incorporation into the Design Guide at the next scheduled revision.

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 the public health and safety. 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 air-cooled graphite 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 guide deals principally with the design and functional requirements of Category VI Air Cooled Graphite Reactor structures, components, and systems.

1.2.1 General Category VI Reactor Characteristics

This guide is specifically applicable to air-cooled graphite reactors with the following general characteristics:

A. An air-cooled graphite reactor is a uranium fueled, graphite moderated and/or reflected reactor that uses air as the cooling medium.

B. An air-cooled graphite reactor normally operates at a low power level such that the energy produced by fission or decay heat is insufficient to require auxiliary cooling. As such, air-cooled graphite reactors do not require primary, secondary, auxiliary, decay heat removal, or emergency core cooling systems. They may utilize an open-loop, once-through air-coolant circuit to transfer heat to the atmosphere to enhance operation and programmatic goals.

Category VI reactors include large-scale facilities that may employ several thousands of kilograms of fuel or small facilities that may contain significant amounts of plutonium in experimental samples. Although these facilities do not require emergency core cooling systems, or decay heat removal systems, they may require confinement and engineered safety feature (ESF) ventilation systems to assure acceptable radiological consequences of postulated accidents. On the

other hand, facilities employing minimal amounts of fuel and insignificant amounts of plutonium in experiment samples may require a minimum of safety-related systems and structures.

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 shut down safely 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 fission products available for release, 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 VI DOE-owned reactors and power reactors, it is possible to simplify the design of the DOE reactor facility while still adequately protecting health and safety. In general, Category VI reactors present less risk to health and safety than do power reactors because of such factors as much lower operating power levels, much lower fission product inventories, remote siting, and reduced dependence on the operation of safety systems required to assure safe reactor shutdown and cooling as well as acceptable consequences of postulated accidents. As a result, Category VI DOE reactor facilities require fewer physical barriers to the release of radioactivity to the environment than power reactors in order to remain within the applicable off-site and on-site dose guidelines specified in the ERDA Manual Chapter 0524, and in 10 CFR 100. For example, some Category VI DOE reactor facilities may 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 the physical isolation of the reactor facility. In addition, inherent safety features such as reduced fission products (due largely to low operating power levels) and enhanced dispersion of released radioactivity (due to remote siting) provide the required reasonable assurance that the DOE-owned 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 corresponding frequency of occurrence and consequence guidelines for normal and accident conditions (the classification of facility operating conditions corresponding to frequency of occurrence is treated in Chapter 15). The radiological consequence guidelines in Table 1.3-1 associated with each facility operating condition are to be interpreted as minimum guidelines to protect health and safety. Each DOE contractor has the responsibility to design or modify a reactor facility such that in addition to meeting the Table 1.3-1 guidelines, the radiological exposures are maintained as low as reasonably

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	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 and ≤300 rem thyroid

*As low as reasonably achievable

achievable. All of the design requirements recommended in this Design Guide have as their basis the above fundamental safety philosophy.

A. Safety Class and Safety Function

Air-cooled graphite reactor facility structures, components, and systems that protect the public health and safety (radiological dose limits within Table 1.3-1) are those required to assure safe shutdown of a reactor and/or to prevent the uncontrolled release of radioactivity to the environment. Structures, components, and systems that protect health and safety are designed to a safety class.

Safety class structures, components, and systems protect health and safety through the implementation and maintenance of certain essential performance characteristics called safety functions. The safety functions are only those essential performance characteristics that are necessary to protect health and safety. Because of the large complexity and varied designs of DOE-owned reactors, it is not within the scope of this Design Guide to specify only those that are necessary for a specific reactor to protect health and safety. The safety functions in this guide are the customary performance requirements that have generally been accepted as the result of industry experience.

The implementation and maintenance of the safety functions is achieved through the design of components, systems, and structures to meet quality standards 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 the three nuclear safety classes (Safety Classes 1, 2 and 3) and a non-safety class (non-nuclear). The design guidelines presented in this Design Guide for determination of an appropriate safety class and other design features are those that are generally considered necessary to implement and maintain the safety functions. It is not within the scope of this Design Guide to identify only those guidelines that are applicable to a specific reactor 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 guide identifies criteria for designating the safety class of structures, components, and systems. These criteria basically provide for a designation of a minimum safety class and, beyond this, general guidelines to be used toward the determination of a final safety class.

1. Minimum Safety Class

The criterion for determining the minimum safety class is that provided by USNRC Regulatory Guide 1.26. If the conservatively calculated potential off-site doses due to a postulated failure of the system can exceed 0.5 rem to the

whole body, the system must be designed to a safety class. If the conservatively calculated potential off-site doses due to a postulated failure of the system do not exceed 0.5 rem, the system may be non-nuclear. Additional guidelines for determining minimum safety class are found in Section 3.2 and in Chapters 4 to 12.

2. Final Safety Class

In addition to this definitive guidance for the minimum safety class requirements, this Design Guide presents general guidance for the determination of a final safety class. The accident analysis performed during the preparation of the Safety Analysis Report (PSAR and/or FSAR, see Chapter 15) results in design specifications such that the limiting consequences of Table 1.3-1 are not exceeded. In the process of arriving at the design specifications, it may be necessary to designate a higher safety classification than that imposed by the minimum safety class criteria presented in Chapter 3. For example, to meet the ERDAM 0524 on-site dose limits for normal and upset conditions, it may be necessary to specify a higher safety class than the minimum recommended in this guide. Such a selection process for a higher safety classification is contingent upon many detailed and complex considerations which are generally applicable only to a particular facility. The Design Guide provides the customary "safety functions" of a system and "design guidelines" which, if adopted, generally result in the maintenance of the systems associated "safety function." These guidelines aid in the determination of the final safety class to which a system is built, but such guidelines cannot be definitive. It is the responsibility of the DOE contractor to investigate the system for the particular site in question together with the recommendations of this Design Guide to determine the final safety class of a system in cases where the minimum recommended in this guide is insufficient for either safety reasons (limit consequences of Table 1.3-1) or for programmatic reasons self-imposed at the particular site.

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 air-cooled graphite reactors 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 air-cooled graphite reactors. Where necessary, design guidelines provide interpretations of the standards, codes, and guides in order to facilitate their application to air-cooled graphite reactors. When existing criteria do not provide complete design guidance, the design guidelines presented in this document serve as interim criteria 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 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 air-cooled graphite reactor or modifying an existing facility. As illustrated in Figure 1.3-1, the design of any system should consider the guidance provided in Chapters 3 and 15, 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 VI air-cooled graphite reactors and also references the Design Guide chapter or section that provides the detailed system design information. A comparable listing for Category VI air-cooled graphite reactors is also found in Table 3.2-2, which also summarizes the principal design codes, safety class, quality group, seismic category, and quality assurance requirements generally applicable to the design of typical Category VI reactor systems. Because of the significant differences in programmatic roles, siting, and power levels among Category VI reactors, it is anticipated that the features of an individual facility may vary substantially from the listings in Tables 1.3-2 or 3.2-2 while still adequately protecting health and safety.

The detailed design guidelines recommended for each system or structure are presented in an individual chapter or section of this Design Guide (Chapters 4 to 12). The following information is included:

- A. Description of the system safety function.
- B. Definition of the system.
- C. Diagrams that provide a guide to the use of the chapter or section.

D. Description of the rationale for determining the need for a system and for determining the system minimum safety class. At the discretion of the designer, a higher safety class may be used in all applications.

Figure 1.3-1. Design guide organization.

Table 1.3-2
Typical Systems for Category VI Reactors ⁽¹⁾

SYSTEM	DESIGN GUIDE SECTION
REACTOR CORE ASSEMBLY	4.1
REACTIVITY CONTROL SYSTEM	4.2
REACTOR BUILDING ⁽²⁾	6.1
ENGINEERED SAFETY FEATURE VENTILATION SYSTEM ⁽³⁾	6.2
PROTECTION SYSTEM ⁽³⁾	7.1
NON-SAFETY INSTRUMENTATION AND CONTROL SYSTEM	7.2
CLASS 1E ELECTRIC SYSTEM ⁽³⁾	8.1
NON-CLASS 1E ELECTRIC SYSTEM	8.2
FUEL HANDLING AND STORAGE SYSTEM	9.1
FIRE PROTECTION SYSTEM	9.2
EXPERIMENTAL FACILITIES	10
RADIOACTIVE WASTE SYSTEM	11
RADIATION PROTECTION SYSTEM	12

- ⁽¹⁾ The systems listed in this table are those which may typically be found at a Category VI Air Cooled Graphite Reactor. The systems listing should not be interpreted as mandatory minimum systems requirements, but rather as a summary of the principal Design Guide Chapters and Sections of interest to the designer.
- ⁽²⁾ Generally, except for facilities using significant amounts of plutonium in experiment samples, this reactor building need not be capable of controlling the release of radioactivity to the environment.
- ⁽³⁾ Generally, safety class systems are not required for facilities with minimal fission product inventories and insignificant amounts of plutonium.

E. General Design Guidelines. This section provides basic guidance for the designer. Where existing codes, standards, and guides adequately address detailed design, functional, or procedural requirements, this section serves to highlight fundamental design criteria. Where necessary, interpretations are provided of the requirements of existing codes, standards, and guides in order to facilitate their application to air-cooled graphite reactors. If no applicable codes, standards, or guides have been developed, this section provides interim design guidance. In any case, the requirements of this section should be met by the designer to the extent necessary to support the associated safety functions. While adherence to the design guidelines is generally required to support the safety functions, any given facility, because of differences in siting, design, etc., may deviate significantly from the guides and still adequately protect the health and safety.

F. Listings of applicable 10 CFR 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, Branch Technical Positions, IAEA Safety Guides, and other general references.

H. Listing 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 VI 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 Guides and for the 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 assigned to 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 (normal, upset, emergency, and faulted) based on frequency of occurrence and severity of the event, for the purpose of determining consequence guidelines applicable to the particular event.

I. Safety Class

The grouping of reactor facility structures, components, and systems 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 of a reactor and/or to prevent 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 50.36a, 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).

CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY, AND DEMOGRAPHY

2.1.1 Guidelines for Facility Siting

A. The size of the facility site and the location of the facility within the site boundary should provide assurance that the guidelines of Chapter 15 will not be exceeded for all postulated accidents.

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 facility.

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 require the analysis of potential on-site doses from postulated accidents. Maximum dose guidelines for persons at the exclusion area boundary are defined in Chapter 15.

D. The 10 CFR 100 guidance concerning the "Low Population Zone" surrounding a facility and the "Population Center Distance" should be considered in siting a facility. Maximum accident dose guidelines for persons at the outer boundary of the "Low Population Zone" are defined in Chapter 15. In the event that the maximum conservatively calculated dose resulting from postulated accidents does not exceed 0.5 rem at the "Exclusion Area Boundary," determination of the "Low Population Zone" and "Population Center Distance," and resulting potential doses in these zones may not be necessary.

2.1.2 Applicable 10 CFR Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the facility 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 facility 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 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"
2. USNRC Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites"
3. USNRC Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations"
4. USNRC Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations"
5. 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-S5, Draft, "Man-Induced Events Related to Nuclear Power Plant Siting," *ibid*.

B. Pending Industry Standards and Codes

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

2. ANSI N635, Draft, (ANS 2.12) "Guideline for Combining Natural and External Man-Made Hazards at Power Reactor Sites"

2.1.5 Topics for Future Standards Development

None

2.2 METEOROLOGY

2.2.1 Guidelines for Facility Siting

A. The facility 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 exist, an on-site meteorological program should be established using suitably accurate and reliable meteorological sensors and a suitable system for data recovery. A general site meteorology program which also provides data for other facilities on site is usually satisfactory.

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 design basis calculations.

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

2.2.2 Applicable 10 CFR Siting Criteria

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

A. 10 CFR 51, "Licensing and Regulatory Policy and Procedures 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 facility 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 1.23, "Onsite Meteorological Programs"

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

3. 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, "Onsite 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. G.C. Holzworth, "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. D.H. Slade (Ed.), "Meteorology and Atomic Energy 1968," TID-24190, Division of Technical Information, USAEC, 1968
12. G.R. Yanskey, E. H. Markee, and A. P. Richter, "Climatology of the National Reactor Testing Station," IDO-12048, Idaho Operations Office, USAEC, 1966

13. J.F. Sagendorf, "A Program for Evaluating Atmospheric Dispersion from a Nuclear Power Station, "Technical Memo ERL ARL-42, NOAA, 1974

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

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

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

B. Pending Industry Standards and Codes

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

2. ANSI N179, 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 Releases 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 Facility Siting

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

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

B. Category VI reactors do not generally require safety class cooling water systems. Water sources, thus, do not require protection against the effects of severe natural phenomena

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

D. The need for and the extent of procedures and measures to protect other groundwater users should be commensurate with the potential releases of radioactivity or toxic material from a facility.

2.3.2 Applicable 10 CFR Siting Criteria

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

A. Industry Standards and Codes

1. ANSI N170-1976, "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 1.135, "Normal Water Level and Discharge at Nuclear Power Plants"

4. 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, "Potential 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 2.4.7, "Ice Effects," *ibid.*
8. USNRC Standard Review Plan 2.4.10, "Flooding Protection Requirements," *ibid.*
9. USNRC Standard Review Plan 2.4.13, "Groundwater," *ibid.*
10. USNRC Standard Review Plan 2.4.14, "Technical Specifications and Emergency Operation Requirements," *ibid.*
11. WASH 1294, "General Environmental Siting Guides for Nuclear Power Plants: Topics and Bases," USAEC, December 1973
12. Reports of the Corps of Engineers, U.S. Army (These reports address many specific hydrology-related topics).
13. 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).

B. Pending Industry Standards and Codes

1. ANSI N515, 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 Facility Siting

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

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

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

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

B. The seismic and geologic siting guidelines of 10 CFR 100, Appendix A, should be considered in the site evaluation.

2.4.2 Applicable 10 CFR Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the facility 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 facility 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 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. G.V. Cohee, et al., "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, "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 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

Air-cooled graphite facility structures, components, and systems required for assuring safe shutdown of a reactor and/or for preventing the uncontrolled release of radioactivity to the environment 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 class systems, and to allow credit to be taken for their capabilities in a conservative safety analysis, minimum requirements are imposed in areas such as design, performance, operation, and quality assurance. This chapter presents general guidance pertinent to the design of safety class structures, components, and systems. 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 the unique design and operating characteristics of most Category VI 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 VI reactor safety systems need 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} per year.

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

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).

example, a Category VI reactor may only be "at risk" to some severe natural phenomena (e.g., tornado-generated missiles) during operation. On the other hand, the same facility may be "at risk" to flooding any time there is a 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 VI 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 VI 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 commercial power reactors, which may be designed to continue operating during some severe natural phenomena in order to supply electric power, air-cooled graphite reactor facilities may be operated on a more flexible schedule that permits shutting down the reactor when the facility is threatened by impending severe natural phenomena. At these facilities, some safety class systems required during reactor operation (e.g., protection system) may not require protection from foreseeable severe natural phenomena if reactor shutdown is required in advance of the severe natural phenomena and the safety class systems are then no longer required for safety. Other safety class systems that are required for maintaining the reactor in a safe shutdown condition or for preventing an uncontrolled release of radioactivity to the environment 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 additional safety class systems against the effects of the severe natural phenomena. Subsequent facility operation would, of course, be preceded by the satisfactory completion of repairs and retesting of unprotected safety class components, systems, or structures that may have been damaged by the severe natural phenomena.

Facility 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 to maintain radiological releases within the guideline values of Chapter 15 for the associated frequency of occurrence. Guidelines defining the spectrum of earthquakes to be considered in Category VI reactor facility design are included in Section 3.2.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

3.2.1 Seismic Classification

Facility structures, components, and systems required to assure safe shutdown of a reactor and/or to prevent uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15 are designated Seismic Category I, and are specifically designed to withstand, without loss of safety function, 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 facility 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 VI 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 VI 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 VI reactor facility, the following examples should be considered:

A. Consider a hypothetical facility that is "at risk" only during operations. On the basis of the expected reactor operating schedule, it is determined that this corresponds to an "at risk" period of approximately 10^{-2} per year. The seismic design of this facility may be based on a DBE with a probability of occurrence in the range of 10^{-4} to 10^{-5} per year. This would be a more likely earthquake than the SSE and, thus, its magnitude should be smaller 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. Some Category VI 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. because of a significant decay heat level and fission product inventory or because of 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 VI 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 VI 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 VI reactor design does, however, allow credit for inherent facility safety features while still satisfying the overall safety objective for an individual reactor facility; namely, to ensure that there is $< 10^{-6}$ per year chance of an accident with dose consequences in excess of the 10 CFR 100 guidelines.

Category VI reactors generally do not require the integrity of the reactor building in order to maintain doses within the applicable guideline; therefore, the reactor building may not need to be Seismic Category I. The reactor core and the fuel in the storage 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. Experimental facilities that present potential radiological consequences exceeding the applicable guidelines should be provided with appropriate Seismic Category I physical protection.

The interfaces between Seismic Category I and Non-Seismic Category I structures and facility equipment should be designed for the dynamic loads and displacements produced by equipment on both sides of the interface. In general, the failure of non-seismic category systems, components, or structures should not impair the integrity of Seismic Category I systems and components from performing their safety function when required. Seismic design guidelines are discussed in Section 3.7.

3.2.2 Classification of Electrical Systems and Components

Electrical components and systems required for preventing an uncontrolled release of radioactivity to the environment and maintaining safe shutdown are designated Class IE and 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 less stringent design and QA requirements.

Category VI reactors do not generally require Class IE electric power systems to safely shut down and maintain a safe shutdown condition. 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

Safety class systems 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 (or 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 a failure of the system can exceed 0.5 rem to the whole body. If these doses do not exceed 0.5 rem, a safety class system is not required, and the system may be designated non-nuclear. Table 3.2-1 presents general guidelines for determining the minimum safety class of Category VI reactor facility components, systems, and structures. 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 Category VI reactors. Table 3.2-2 is not to be interpreted as a listing of required systems for Category VI reactors. It is rather a listing of systems that would be found at a hypothetical "typical facility." Because of the significant differences in programmatic roles, siting, and power levels among Category VI reactors, it is anticipated that the features of an individual facility may vary substantially from the listings in Table 3.2-2 while still adequately protecting health and safety.

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 the 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 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.

Category VI reactors do not generally require equipment designed to the ASME Boiler and Pressure Vessel Code. If a designer at a particular Category VI reactor facility should determine that some systems or components should be designed to the requirements of the ASME code (e.g., experimental facility test loops and equipment), the Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors, should be consulted.

TABLE 3.2-1

GENERAL SAFETY CLASS DEFINITIONS (1) (2)

Safety Class 1 (Quality Group A)

Air cooled graphite reactor facilities generally do not require SC-1 structures, components and systems.

Safety Class 2 (Quality Group B)

Those systems, the failure of which may result in the radiological consequences of accidents being in excess of the 10 CFR 100 guidelines. (May include ESF ventilation system, containment/confinement systems, experimental facility test loops.)

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 (see Chapter 15) doses in excess of 0.5 rem to the whole body or its equivalent to any part of the body, but less than 10 CFR 100 dose guidelines.

- Note: (1) A Non-Nuclear (NN) designation (Quality Group D) may be designated to those components, systems or structures that contain or may contain radioactive material that do not have the potential for causing off-site (see Chapter 15) doses in excess of 0.5 rem to the whole body or its equivalent to any part of the body.
- (2) General guidance for determining the final safety class are presented in Section 3.2 and in Chapter 15.

Table 3.2-2

Typical Equipment Classification for Category VI Reactors

SYSTEM OR COMPONENT	DESIGN GUIDE SECTION	MINIMUM SAFETY CLASS	QUALITY GROUP	SEISMIC CATEGORY	PRINCIPAL DESIGN CODE OR STANDARD
REACTOR CORE ASSEMBLY	4.1	-	-	I	-
REACTIVITY CONTROL SYSTEM	4.2	(a)	-	I	-
REACTOR BUILDING (b)	6.1	(c)	-	(c)	-
PROTECTION SYSTEM (d)	7.1	-	-	-	IEEE 279
NON-SAFETY INSTRUMENTATION AND CONTROL SYSTEM	7.2	(f)	D	-	NEMA ICS-1970
NON-IE ELECTRIC POWER SYSTEM	8.2	(f)	D	-	ANSI C-1971, NEMA MG 1-1972
FUEL HANDLING AND STORAGE SYSTEM	9.1	(e)	(e)	(e)	-
FIRE PROTECTION SYSTEM	9.2	(f)	D	-	NFPA 802-1974
EXPERIMENTAL FACILITIES	10	(g)	(g)	(g)	-
RADIOACTIVE WASTE SYSTEM	11	(f)	D	-	ASME B&PV CODE, SECTION III
RADIATION PROTECTION SYSTEM	12	(f)	D	-	-

(a) See guidelines in Section 4.2.

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

(c) See guidelines in Section 6.1.

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

(e) See guidelines in Section 9.1.

(f) Not required to be designed to safety class standards. May be designed to Non-Nuclear standards.

(g) Experimental facilities may interface with other facilities (e.g., the reactor core assembly, instrumentation, control, and electrical systems). As a minimum, the experimental facilities should be designed to safety class seismic category and QA standards that are compatible with the design requirements of interfacing systems and which additionally considers the potential hazards associated with the experimental facility.

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, 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 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., "Specification 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 1, 2, and 3 systems that are not subject to the ASME code 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 10 CFR 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"

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

B. RDT Standards

None

C. Guides

None

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

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

3.2.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful 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, the following wind loading design guidelines should be considered.

3.3.1 Design Guidelines

A. Structures, components, and systems should be designed to withstand or to be protected from the effects of wind and tornado loadings if they must function during or after such natural phenomena in order to safely shut down the reactor and/or to maintain radiological releases within the guideline values of Chapter 15. As a minimum, the 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.

B. Some safety class systems may not require protection from wind loadings or tornados if the reactor can be shut down in advance of the severe meteorological conditions and the systems' safety functions are no longer required.

C. 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 at nuclear power plants, except those on exposed coastal locations. The loading 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 10 CFR General Design Criterion

It is recommended that the following 10 CFR 50, Appendix A, General Design Criterion be considered in designing a 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 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, Proposed (ANS 2.3), "Guidelines for Estimating Tornado, Hurricane, and Other Extreme Wind Parameters at Power Reactor Sites"

2. ANSI N171, "Tornado Design Criteria"

3.3.5 Topics for Future Standards Development

None

3.4 WATER LEVEL (FLOOD) DESIGN

For safety class systems identified in Section 3.2, the following flood protection design guidelines should be considered.

3.4.1 Flood Design Guidelines

A. Structures, components, and systems should be designed to withstand or be protected from the effects of flooding if they must function during or after such natural phenomena to safely shut down the reactor and/or to maintain radiological releases to within the guideline values of Chapter 15. As a minimum, the reactor and the fuel in storage should be protected against flooding. Protection requirements for other facility systems should be based on the above general guidelines.

B. The magnitude of the design basis flood, including attendant wind-generated wave activity for which the 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.

C. Methods of flood protection to be considered should include:

1. Dry Site
2. Exterior Barriers
 - a. Levee
 - b. Seawall or Floodwall
 - c. Bulkhead
 - d. Revetment
 - e. Breakwater

3.4.2 Applicable 10 CFR General Design Criterion

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

- A. Industry Standards and Codes

1. ANSI N170-1976/ANS-2.8, "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 that are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 3.4.1, "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

Structures, components, and systems required for maintaining a reactor in a safe shutdown condition 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 potential missiles. Missiles generated within the facility are usually generated by equipment failures or accidents at the facility.

The probability of occurrence of missiles capable of damaging safety class facility systems, components, or structures and resulting in exposures comparable to the 10 CFR 100 guidelines should not exceed the probability guidelines for design basis accidents discussed in Chapter 15.

3.5.1 Design Guidelines

3.5.1.1 Missiles Generated at the Facility

A. Among the facility-generated missiles to be considered as potential hazards are those arising from failure of experimental facilities, highspeed rotating machinery, or containers of explosive fluids. Examples of potential missiles are valve bonnets, bolts, instrument wells, gas cylinders, and fan blades.

B. 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 other protective structures should be provided to ensure that safety class systems can withstand the effects of potential missiles. Appropriate credit may be taken for the protection afforded by a biological shield surrounding the reactor and some other reactor system components. The biological shield may effectively limit the potential effects of some facility-generated missiles.

3.5.1.2 Missiles Generated by Natural Phenomena

A. Tornadoes, hurricanes, earthquakes, and floods are capable of generating missiles which have the potential for disabling safety class systems. Safety class systems 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 should prevent potential externally generated missiles from contacting the reactor.

B. The design basis missiles at a particular site should be determined on the basis of the frequency of specific severe natural phenomena and the potential consequences of missile strikes on safety class systems of the facility.

C. Generally, tornado-generated missiles are considered the most limiting. A representative spectrum of tornado-generated missiles is included in WASH 1361.

3.5.1.3 Other Externally Generated Missiles

A. Aircraft crashes and explosions on nearby transportation routes may generate missiles from which safety class systems at a 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).

B. 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 overpressure that exceeds the equivalent tornado dynamic wind pressure.

C. The assessment of aircraft hazards at the site should also be performed and the risk found to be acceptably low.

3.5.2 Applicable 10 CFR General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criteria be considered in designing a 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"

3.5.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in designing a 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"

3.5.4 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 3.5.1.1, "Internally Generated Missiles (Outside Containment)," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 3.5.1.2, "Internally Generated Missiles (Inside Containment)," *ibid.*
3. USNRC Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," *ibid.*
4. USNRC Standard Review Plan 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," *ibid.*
5. USNRC Standard Review Plan 3.5.1.6, "Aircraft Hazards," *ibid.*
6. USNRC Standard Review Plan 3.5.2, "Structures, Systems, and Components to be Protected from Externally Generated Missiles," *ibid.*
7. USNRC Standard Review Plan 3.5.3, "Barrier Design Procedures," *ibid.*
8. Branch Technical Position AAB 3-2, "Tornado Design Classification (Appended to SRP 3.5.1.4)," *ibid.*
9. WASH 1361, "Safety-Related Site Parameters for Nuclear Power Plants," January 1975
10. Department of the Army Technical Manual TM5-1300, "Structures to Resist the Effects of Accidental Explosions," June 1969

B. Pending Industry Standards and Codes

None

3.5.5 Topics for Future Standards Development

None

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

Air-cooled graphite reactors generally do not use piping systems that require the consideration of dynamic effects resulting from postulated rupture of piping. Should experimental facilities utilize high energy systems that may require consideration of the dynamic effects associated with postulated pipe ruptures, the user should consult the Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors.

3.7 SEISMIC DESIGN

For safety class systems designated Seismic Category I in Section 3.2, the following seismic design guidelines should be considered.

3.7.1 Seismic Input

The seismic input to the facility design should be based on a detailed seismic investigation of the facility 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 past earthquakes. The degree of scaling necessary is based on the relative intensity of the DBE (or the SSE, if appropriate) at the actual facility site. The response spectra recommended in USNRC Regulatory Guide 1.60 corresponds 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, components, and systems should conform with or be more conservative than those 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. Dynamic analysis of systems, unless the equivalent static load method can be justified, based on simple system configuration and conservative analytical results.

B. Natural frequency and response load analysis, including the time-history of acceleration or response spectrum at the major facility equipment elevations and points of support.

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 shutdown 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, and/or repairs, as appropriate.

3.7.4 Applicable 10 CFR General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criteria and provisions of 10 CFR 100 be considered in the seismic design of air-cooled graphite 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 Category VI 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 Responses 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 of 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.D., et al., "Guidelines for Developing Design Earthquake Re-
sponse Spectra," CERL-TR-M114, Army Construction Engineering Research Labora-
tory, 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, Draft, "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 Nu-
clear 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. ANS 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

For structures designated Seismic Category I, the seismic design guidelines of Section 3.7 and the following design guidelines for Category VI structures should be considered.

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 limit or control the release of radioactivity

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 facility reactor building are presented in Section 6.1. Loading combinations and the design and analytical procedures to be considered in the design of these Seismic Category I structures are discussed in several standards and codes listed in the following subsections. The loads normally applicable to steel or concrete facility reactor buildings include the following:

- A. Those loads encountered during preoperational testing.
- B. Those loads encountered during normal facility startup, operation, and shutdown.
- C. Those loads to be sustained during severe environmental conditions, including those induced by the design wind specified for the facility 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 facility site.
- E. Those loads to be sustained after abnormal facility conditions.

In general, these structures should be designed to facilitate periodic testing and inspection throughout the life of the facility in order to verify that leak rates, 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 have several safety-related functions for which their structural integrity is important. By providing support for experiments during normal operations and during seismic disturbances, they should help prevent the occurrence of an unplanned reactivity transient caused by movement of an experiment near the reactor. Loads normally applicable to facility building internal structures are similar to those considered in Subsection 3.8.1 for the reactor building.

Air-cooled graphite reactor building safety class 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 functions when required.

3.8.3 Other Seismic Category I Structures

Structures other than a Category I reactor building and its internal structures may have safety functions associated with protection of safety class components or systems such as a control room or fuel storage building. Design requirements for these structures are specified in ANSI A89.1-1972 and in AISC

steel specifications. Loads normally applicable to these Seismic Category I structures are similar to those considered in Subsections 3.8.1 for the reactor building.

Other Seismic Category I structures should be designed to facilitate periodic testing and inservice inspection as required to assure their continued integrity and ability to perform their safety functions when required.

3.8.4 Applicable 10 CFR 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 facility buildings (including confinements or safety-related 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 6.1.

- 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 facility buildings (including confinements or safety-related 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 6.1.

A. Industry Standards and Codes

1. ANSI A 58.1-1972, "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 301-72), "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," *ibid.*
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 are defined in the ASME Boiler and Pressure Vessel Code. 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. These factors affect the potential consequences of accidents and the frequency of their occurrence. By design, the more frequent events should have lesser radiological consequences and should be accommodated within lower stress limits than the less frequent accidents. The spectrum of postulated accidents to be considered in the design of air-cooled graphite reactor facilities is listed in Chapter 15.

3.9.1 Safety Class Mechanical Components

Category VI reactors generally require few safety class mechanical components. Of those that are required, most are not covered by the ASME Boiler and Pressure Vessel Code. If it is determined that some systems or components of a particular Category VI reactor facility should be designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, the Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors, should be consulted for the appropriate design guidelines.

Safety class mechanical components at Category VI reactor facilities may include the following:

- A. Fuel assemblies
- B. Other portions of the reactor structure (e.g., core support, control rod passages).
- C. Mechanical portions of the reactivity control system.

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 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 to ensure the safe shutdown of the reactor and/or to prevent an uncontrolled release of radioactivity with dose consequences in excess of the guideline in Chapter 15.

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 FACILITY OPERATION	STARTUP, SHUT-DOWN, REFUELING.
UPSET	INCIDENTS OR OCCURRENCES OF MODERATE FREQUENCY WHICH MAY OCCUR DURING A CALENDAR YEAR AT A PARTICULAR FACILITY.	LOSS OF ELECTRIC POWER, PARTIAL LOSS OF COOLING AIR FLOW, INADVERTENT REACTOR SHUTDOWN.
EMERGENCY	INFREQUENT INCIDENTS OR OCCURRENCES WHICH MAY OCCUR DURING THE LIFETIME OF A FACILITY.	ROD WITHDRAWAL ACCIDENT, REACTIVITY EXCURSION.
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

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 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 operating 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 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, by 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.

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"

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.

3.9.5 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 of Section 3.2 that instrumentation and electric equipment should be Seismic Category I, the following seismic qualification guidelines are applicable:

3.10.1 Seismic Qualification

Systems and components that must retain structural integrity, remain leak tight or continue to function in the event of an earthquake in order to prevent an uncontrolled release of radioactivity to the environment are designated Seismic Category I. Qualification of Category I instrumentation and electrical equipment is through the demonstration of operability by test, detailed stress and deformation analysis or a combination of these methods. 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.

D. Supports for Seismic Category I instrumentation and electrical equipment should be tested with the equipment in place or with simulated equipment.

Qualification by analysis method is not recommended for complex equipment that cannot be modeled to correctly predict its response. Analysis without testing may be acceptable if structural integrity alone can assure the design-intended function.

3.10.2 Applicable 10 CFR General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criterion be considered in the qualification of Seismic 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 qualification of Seismic Category I 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, 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

Typical air-cooled graphite reactor mechanical and electrical equipment that should consider the following environmental guidelines are: control rods and their scram mechanisms, portions of the protection systems, and contaminated air exhaust filters.

3.11.1 Environmental Design Guidelines

All items of safety class mechanical and electrical equipment should be designed to be capable of performing their design safety functions under all normal and accident environmental conditions. The "normal and accident environmental conditions" are deemed to include all environmental conditions that may result from any normal or abnormal mode of facility operation, design basis events, and system tests.

The general requirements for environmental design and qualification are as follows:

- A. The equipment should be designed to have the capability of performing design safety functions under all normal and accident environments, where the specific environments are determined on the basis of the potential spectrum of accidents at a particular facility (see Chapter 15).

Temperature, pressure, and humidity environments should be based on the most severe postulated accident. This is generally the design basis reactivity transient. The postulated environment should reflect the chemical composition of all fluids and additives present in the reactor system or added to the

reactor building environment in the course of an accident. The radiation environment should reflect source terms that are compatible with those postulated to arise from a severe reactivity transient accident.

B. Equipment qualification should provide assurance that safety class systems will be capable of performing required safety functions under accident conditions for at least the period of time that its safety function is required.

C. The equipment environmental capability should be demonstrated by appropriate testing and/or analyses.

3.11.2 Applicable 10 CFR General Design Criteria

It is recommended that the following General Design Criteria and other provisions of 10 CFR 50 be considered in the environmental design and qualification of safety class mechanical and electrical equipment:

- A. GDC 1, "Quality Standards and Records"
- B. GDC 4, "Environmental and Missile Design Basis"
- C. GDC 23, "Protection System Failure Mode"
- D. 10 CFR 50, Appendix B, Section XI, "Test Control"

3.11.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the environmental design and qualification of safety class mechanical and electrical equipment:

A. Industry Standards and Codes

- 1. IEEE 323-1974 (ANSI N41.5), "Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations"
- 2. IEEE 334-1974 (ANSI N41.9), "Standard Type Test for Continuous Duty Class IE Motors for Nuclear Power Generating Stations"
- 3. IEEE 382-1972 (ANSI N41.6), "Trial-Use Guide for Type Test of Class IE Electric Valve Operators for Nuclear Power Generating Stations"
- 4. IEEE 383-1974 (ANSI N41.10-1975), "Standard for Type Test of Class IE Electric Cables Field Splices, and Connections for Nuclear Power Generating Station"
- 5. 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 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants"

2. USNRC Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants"

3. USNRC Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants"

4. USNRC Regulatory Guide 1.131, "Qualification Tests of Electric Cables, Field Splices and Connections for Water-Cooled Nuclear Power Plants"

3.11.4 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 3.11, "Environmental Design of Mechanical and Electrical Equipment," NUREG 75/087, September 1975

2. Branch Technical Position EICSB 23, "Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown," *ibid.*

3. Branch Technical Position MEB 2, "Pump and Valve Operability Program," *ibid.*

B. Pending Industry Standards and Codes

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

2. ANSI N689, Proposed, "Requirements for Qualification of Nuclear Power Plant Safety-Related Equipment"

3.11.5 Topics for Future Standards Development

None

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 the remaining chapters of this Design Guide.

3.12.2 Applicable 10 CFR General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criterion 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 nuclear components:

A. Industry Standards and Codes

- 1. AWS A3.0-76, "Welding Terms and Definitions"
- 2. AWS D1.1-75, "Structural Welding Code"
- 3. AWS D10.9-69, "Standard for Qualification of Welding Procedures and Welders for Piping and Tubing"
- 4. AWS D12.1-1975, "Reinforcing Steel Welding Code"
- 5. ANSI Z49.1-1973, "Safety in Welding and Cutting"

B. RDT Standards

The following RDT standard expands on requirements of the referenced AWS standard (listed in parentheses below). The applicability of the RDT requirements 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 materials in structures, components, and systems that may not be identified in the previously recommended design codes, standards, and guides.

3.13.2 Applicable 10 CFR General Design Criterion

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

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

3.12.3 Standards, Codes, and Guides

It is recommended that, in addition to materials identified in previous standards, codes, and guides, the following be considered in the selection and use of facility materials:

- A. Industry Standards and Codes

1. ANSI N143-1973 (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

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

1. M 16-3T, "HEPA Filter Medium, Glass Fiber" (MIL-F51097 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 List," Underwriters Laboratory, Inc.
2. MIF-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 air-cooled graphite reactor facilities and to the design and operational aspects that may affect decommissioning.

A. Four major alternatives for the retirement of a reactor facility are:

1. Mothballing - The facility is placed in protective storage, generally with all fuel elements, radioactive fluids, and waste removed from the facility. Appropriate monitoring, surveillance, and security programs are required.

2. In-Place Entombment - After removing all fuel elements, radioactive fluids, wastes, and selected components, the remaining highly radioactive components are sealed within a structure integral with the biological shield or other suitable boundary. An appropriate surveillance program is required.

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

4. Conversion to a New Nuclear System - Portions of a reactor 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 practicable (see Chapter 12). Such provisions may include:

1. Provisions for flushing contaminated systems
2. Adequate fixed or movable shielding
3. System design to facilitate the use of remote handling devices
4. Use of ventilation systems to control airborne contamination

C. Procedures for decommissioning should assure that occupational and off-site radiation exposures are maintained as low as practicable (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 10 CFR General Design 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-1917, May 1975

2. AID/NESPO09SR, "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 Research Reactors"

3.14.5 Topics for Future Standards Development

The following topical listing identifies areas in which the development of additional nuclear standards may result in a more consistent approach to air-cooled graphite reactor facility design and modification:

A. Guidelines for reactor facility design to facilitate eventual decommissioning.

CHAPTER 4

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

The safety functions of the reactor core assembly are:

- A. To maintain fuel geometry in order to ensure that the reactor can be safely shut down during normal operating and accident conditions.
- B. To form the primary barrier against the release of fission products from the fuel.

4.1.3 Definitions

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

- A. The Fuel and Fuel Assemblies
- B. Core Support Structures
- C. Thermal Shields
- E. In-Core Experimental Facilities
- F. Moderator and/or Reflector Elements

4.1.4 Design Guidelines

4.1.4.1 Guidelines for the Design of the Core Assembly

A. For normal and upset facility conditions (defined in Table 3.9-1), the reactor should be designed such that fuel-cladding damage or fuel melting is not likely to occur during its design lifetime. 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 conditions.

B. For emergency and faulted facility conditions (defined in Table 3.9-1), the reactor should be designed such that any fuel or cladding damage that occurs does not prevent reactor shutdown.

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

C. The potential effects of experimental facilities associated with the reactor core, including test loops and sample irradiation holes 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 A and B above.

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

E. The design of mechanical components of the core assembly should consider the guidelines in Section 3.9.

F. Instrumentation should be provided to monitor reactor core power over the full range of operation. Protection system (see Section 7.1) inputs should be provided as required to prevent core assembly damage in excess of the guidelines in A and B above except where fuel failure is the initiating event.

G. The reactor core assembly should be protected from the effects of severe natural and man-made phenomena (see Chapter 3). Since Category VI reactors may not use a building containment, the facility design may require additional structures or barriers to assure adequate protection of the core assembly from the effects of the above hazards.

H. Safety class portions of the reactor core assembly should be Seismic Category I (see Chapter 3).

I. Reactor graphite moderators and/or reflectors should be protected from severe oxidation if analyses show this condition to pose a significant hazard.

J. The stored energy in graphite due to irradiation should be considered and appropriate design features and/or operating procedures be incorporated to ensure no significant hazard results from its release.

4.1.4.2 Maintenance, Testing, and Inspection

A. The core assembly should be designed to allow for periodic inspection and maintenance of in-core experimental facilities.

B. Preoperational testing should be performed as per Chapter 14.

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. Reactor Inert Gas System

C. Reactor Cooling System

- D. Protection System
- E. Fuel-Handling and Storage Systems
- F. 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.

4.1.4 Applicable 10 CFR 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's be considered in the design of a reactor core assembly:

- A. GDC 10, "Reactor Design"
- B. GDC 11, "Reactor Inherent Protection"
- C. GDC 12, "Suppression of Reactor Power Oscillations"

4.1.5 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 N147-1973, "Practice for Examination of Fuel Element Cladding, Including the Determination of Mechanical Properties"
 - 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
 - 1. USNRC Regulatory Guide 2.4, "Review of Experiments for Research Reactors"

4.1.6 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 4.2, "Fuel System Design," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 4.3, "Nuclear Design," *ibid.*

B. Pending Industry Standards and Codes"

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

2. ANSI N410, Draft, "Feasibility of Selecting Nuclear Cross Section Data as Standard"

3. ANSI N522, Draft, "Definitions of Reactor Physics Terms and Parameters"

4. ANSI N701, Draft, "Criteria for the Reactor Shutdown Systems of Research Reactors"

5. 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 safety functions of the reactivity control systems are:

A. Control the reactor core reactivity to ensure that the reactor may be safely shut down and that the fuel damage limits of Section 4.1 are not exceeded for any facility condition.

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

B. Limit the damage to the reactor systems that may result from an accident such that their associated safety functions are maintained when required.

4.2.3 Definition

The reactivity control system is defined as the neutron absorbing system which must function in order to assure safe operation and shutdown of the reactor, including control rods and poison shot system.

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 portions (e.g., hydraulic, pneumatic systems) of the reactivity control system should be a minimum Safety Class 3 if their failure prevents safe shutdown of the reactor, or results in consequences in excess of the core damage guidelines of Section 4.1, or the radiological guidelines of Chapter 15. Other mechanical or fluid portions of the reactivity control system may be designed to non-nuclear standards.

B. Electrical portions of the reactivity control system required for safety are Class IE and should consider the guidelines for protection systems and Class IE electric systems discussed in Sections 7.1 and 8.1, respectively. These portions need include only the parts of the reactivity control system required for ensuring safe reactor shutdown and for limiting potential damage to the core assembly to the guideline values in Section 4.1 or radiological guidelines of Chapter 15. Other electrical portions of the reactivity control system may be designed to Non-Class IE standards.

4.2.4.2 Guidelines for the Design of a Reactivity Control System

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

B. Two independent and diverse means of reactivity control should be provided. These may include control rods, poison shot system, movable reflector elements, or other suitable means of reactivity control.

C. Both reactivity control systems should be capable of shutting down an operating reactor; however, only one of the reactivity control systems need be capable of holding the reactor subcritical in a cold condition with any one reactivity control assembly in the position which results in the greatest insertion of positive reactivity.

D. Generally, two reactivity control systems should be capable of reliably controlling reactivity changes under conditions of normal operations, upset conditions, and those emergency conditions that are expected to occur once in the lifetime of the facility. However, inherent reactor safety features, such as a large negative (and, in some facilities, prompt) temperature coefficient and/or

limited excess reactivity, may effectively limit the consequences of postulated reactivity transients that may arise during such conditions thereby providing an equivalent level of safety using only one system for normal operating reactivity control while maintaining the second system available for rapid insertions of negative reactivity when required.

E. Potential accidents that can increase the reactivity of the core and redistribute power, including those arising from the operation or failure of experimental facilities, should be evaluated to determine the response time and reactivity addition rates required for the reactivity control system.

F. The reactor and reactivity control system should accommodate upset conditions (defined in Table 3.9-1) and facilitate a rapid return to normal conditions. Generally, this implies that upset conditions do not require a reactor shutdown by the operator or the protection system.

G. Emergency and faulted events (defined in Table 3.9-1) should be accommodated by the reactor and reactivity control system, and the reactor should be placed in a safe condition.

H. Safety class portions of the reactivity control system should consider:

1. Severe natural and man-made phenomena of Chapter 3 under which they must perform their safety functions.

2. Designation as Seismic Category I as per Section 3.2.

3. Environmental design requirements of Section 3.11.

I. Those portions of the reactivity control system designated Seismic Category I should consider the guidance of Section 3.7 for seismic design and Section 3.10 for seismic qualification.

J. The design of mechanical portions of the reactivity control system should consider the guidelines of Section 3.9.

4.2.4.3 Maintenance, Testing, and Inspection

A. The reactivity control system should be designed to permit periodic functional tests of the entire system, including tests necessary to determine the actual reactivity worth of reactivity control assemblies as necessary to ensure that the system will be capable of completing its safety function.

B. Testing provisions of Sections 7.1 and 8.1 apply to the safety class electrical portions of the reactivity control 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
- E. 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 10 CFR 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 systems:

A. Safety Class Components

1. GDC 10, "Reactor Design"
2. GDC 12, "Suppression of Reactor Power Oscillations"
3. GDC 13, "Instrumentation and Control"
4. GDC 20, "Protection System Functions"
5. GDC 24, "Separation of Protection and Control Functions"
6. GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"
7. GDC 26, "Reactivity Control System Redundancy and Capability"
8. GDC 27, "Combined Reactivity Control Capability"
9. GDC 28, "Reactivity Limits"
10. GDC 29, "Protection Against Anticipated Operational Occurrences"

B. Non-Safety Class Components

The following previously listed GDCs are recommended for consideration in the design of non-safety class reactivity control system components:

GDCs 10, 12, 13, 24 through 29.

4.2.6 Standards, Codes, and Guides

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

A. Industry Standards and Codes

None

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

1. USNRC Standard Review Plan 4.3, "Nuclear Design," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 4.6, "Functional Design of Reactivity Control Systems," *ibid.*

3. Branch Technical Position EICSB 14, "Spurious Withdrawal of Single Control Rods in PWRs" (Appendix to SRP, Chapter 7), *ibid.*

B. Pending Industry Standards and Codes

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

4.2.8 Topics for Future Standards Development

None

CHAPTER 5

REACTOR COOLING SYSTEM

Category VI reactors do not generally require a reactor cooling system. Reactor heat is rejected to the reactor building atmosphere. In some cases, a forced air coolant system may be utilized for programmatic reasons only. The overall ratio of heat removal from the core by the forced air coolant is not crucial since it merely influences the core operating temperature at a given steady-state power level, and the postoperating cooling period. This system is not required for safety.

Facilities that use a forced air coolant system should give appropriate consideration to the release of argon-41. Argon-41 is a short lived radioisotope produced from neutron activation of argon-40 in air.

CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 REACTOR BUILDING

6.1.1 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 confinement, safety-related reactor building or non-safety-related reactor building. Basic guidance relevant to the design of these structures is provided. Figure 6.1-1 is a guide to the use of this section.

6.1.2 Safety Function

A reactor building may have some or all of the safety functions listed below. The relationship between these safety functions and the reactor building types is illustrated in Table 6.1-1.

A. Form a barrier against the uncontrolled release of radioactivity to the environment such that ERDAM Chapter 0524 and 10 CFR 100 guidelines are not exceeded following postulated accidents (see Chapter 15).

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

6.1.3 Definitions

The various types of reactor buildings used with air-cooled graphite reactors are defined below:

A. 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 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 includes those structures necessary for protecting enclosed systems against the effects of severe natural and man-made phenomena.

B. 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

Figure 6.1-1. Guide to the use of section 6.1.

Figure 6.1-1. Guide to the use of section 6.1.

Table 6.1-1

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 AND MAN-MADE PHENOMENA	BUILDING DESIGN CONDITIONS NOT EXCEEDED DURING POSTULATED ACCIDENTS	
CONFINEMENT	YES	YES	YES	1. DIRECTED FILTER EXHAUST STREAM 2. SEISMIC CATEGORY I STRUCTURE
SAFETY RELATED REACTOR BUILDING	NO	YES	YES	SEISMIC CATEGORY I STRUCTURE
NON-SAFETY RELATED REACTOR BUILDING	NO	NO	NO	INDUSTRIAL TYPE BUILDING

severe natural and man-made phenomena, but which are not intended for controlling the release of radioactivity to the environment.

C. 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 and man-made phenomena.

6.1.4 Design Guidelines

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

The following guidelines describe the basis for selecting a confinement, safety-related reactor building or non-safety-related reactor building to house a reactor and associated systems. The need for a 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 and man-made phenomena, will be based on the protection requirements of the enclosed systems. The minimum safety class guidelines for the reactor building are listed in Table 6.1-2. Design guidelines are presented in subsections 6.1.4.2 through 6.1.4.6.

A. Confinement

The reactor building should be designed as a 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 confinement should be based on the degree of off-site dose or effluent concentration reduction required. A confinement provides protection to enclosed systems and components against the effects of severe natural and man-made phenomena.

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 does not require confinement capability in order to operate within the appropriate off-site dose and effluent concentration guidelines. If enclosed systems or components require protection against the effects of severe natural and man-made phenomena, the reactor building is safety-related 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 and, therefore, requires that enclosed safety class systems or components be provided with other suitable protection.

Table 6.1-2

Minimum Safety Class Recommendations for Reactor Buildings

REACTOR BUILDING TYPE	POTENTIAL RADIOLOGICAL CONSEQUENCES OF ACCIDENTS		
	THE MAXIMUM OFF-SITE RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS WOULD EXCEED 10CFR100 GUIDELINES IF THE REACTOR BUILDING WERE NOT PROVIDED	THE MAXIMUM OFF-SITE RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS WOULD BE >0.5 REM BUT ≤ 10CFR100 GUIDELINES IF THE REACTOR BUILDING WERE NOT PROVIDED	THE MAXIMUM OFF-SITE RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS WOULD BE ≤0.5 REM IF THE REACTOR BUILDING WERE NOT PROVIDED
CONFINEMENT	SC-2 (1)	SC-3 (2)	-
SAFETY RELATED REACTOR BUILDING	-	-	SC-3 (3)
NON-SAFETY RELATED REACTOR BUILDING	-	-	NN (4)

NOTE (1) SC-2 BASED ON GUIDELINES OF USNRC REGULATORY GUIDE 1.26 FOR QUALITY GROUP B.

(2) SC-3 BASED ON DOSE GUIDELINES OF REGULATORY GUIDE 1.26 FOR QUALITY GROUP C.

(3) SC-3 BASED ON NEED TO PROVIDE PROTECTION TO ENCLOSED STRUCTURES.

(4) NN BASED ON GUIDELINES OF REGULATORY GUIDE 1.26 FOR QUALITY GROUP D OR NO REQUIREMENT TO PROVIDE PROTECTION TO ENCLOSED STRUCTURES.

6.1.4.2 Guidelines for the Design of a Confinement and Associated Systems

A. The Confinement Structure

1. The design pressure and temperature of the reactor confinement structure should be determined considering the effects of the following:

a. Postulated accidents including resulting residual heat, fission product decay heat, and stored energy in graphite.

b. Static heat sources and sinks.

c. Transport of stored energy from heat sources to the confinement.

d. Such items as graphite oxidation, combustible gas reactions and other possible energy sources, and uncertainties that may necessitate additional design margin.

2. The confinement system and associated systems should be capable of performing the required safety functions on off-site or on-site electric power (assuming one is unavailable).

3. The confinement and associated systems required for safety are Seismic Category I structures (see Chapter 3).

4. The confinement structure should provide protection to enclosed structures from the effects of severe natural and man-made phenomena (see Chapter 3).

5. Provisions should be included for preventing the accumulation of potentially explosive mixtures of combustible gas that may be generated following postulated accidents.

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

7. Filter systems installed in the confinement exhaust path should incorporate sufficient redundancy to ensure that dose guidelines for the appropriate facility operating conditions (see Chapter 15) are not exceeded following a single active failure in the exhaust system.

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

9. A failure of a single active component in the exhaust system should not result in back leakage of radioactivity through confinement air intakes unless that flow path is adequately filtered and the radiological consequences are acceptable.

10. Elevated confinement exhaust locations are preferred in order to limit on-site doses and to reduce the off-site doses due to enhanced atmospheric dispersion.

B. The Confinement Isolation Systems

A confinement isolation system is generally not required. The spectrum of postulated accidents (see Chapter 15) does not cause any significant over-pressure in the reactor building, therefore, negligible leakage will occur through confinement penetrations.

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

A. The safety-related reactor building guidelines are similar to those specified in Subsection 6.1.4.2 for confinement design in the following areas:

1. Potential sources of energy release to the reactor building
2. Seismic classification
3. Protection against severe natural and man-made phenomena
4. Safety class of associated systems
5. Prevention of accumulations of combustible gas

B. No isolation system is required to control the release of radioactivity to the environment.

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

A. Provisions should be included for preventing the accumulation of potentially explosive mixtures of combustible gas that may be generated following postulated accidents.

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

C. No isolation system is required to control the release of radioactivity to the environment.

6.1.4.5 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 shut down.

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

6.1.4.6 Interfaces With Other Systems

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

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

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

6.1.5 Applicable 10 CFR 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 provisions of 10 CFR 50 be considered in the design of reactor buildings:

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

B. GDC 64, "Monitoring Radioactivity Releases"

6.1.6 Standards, Codes, and Guides

In addition to the general standards, codes, and guides listed in Chapter 3, it is recommended that the following 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, "Standard for Test Performance of High Efficiency Particulate Air Filter Units"

2. Safety-Related Reactor Buildings

It is recommended that the following previously listed industry standard be considered in the design of a safety-related reactor building:

- a. ANSI A158.1-1970

3. Non-Safety-Related Reactor Buildings

It is recommended that the following previously listed 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 for 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. "Specifications 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. E9-1T, "HEPA Filters"

C. Guides

1. 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"
2. 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"
3. USNRC Regulatory Guide 3.2, "Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles"

6.1.7 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 6.2.4, "Containment Isolation System," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 6.2.6, "Containment Leakage Testing," *ibid.*
3. USNRC Standard Review Plan 6.5.1, "ESF Filter Systems," *ibid.*
4. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.
5. ORNL-NSIC-26, "Testing of Containment Systems Used with Light Water-Cooled Power Reactors" Oak Ridge National Laboratory, August 1968
6. Burchsted, C.A. et al., "Nuclear Air Cleaning Handbook," ERDA 76-21, Oak Ridge National Laboratory, March 31, 1976
7. Griwatz, G.H., Friel, J.U., and Bicehouse, J.L., "Entrained Moisture Separators for Fine (1-10) Water Air-Steam Service: Their Performance, Development, and Status," MSAR71-45, Mine Safety Appliances Research Corporation
8. MIL-F-51068D, "Filter Particulate High Efficiency Fire Resistant"
9. "Leakage Characteristics of Openings for Reactor Housing Components" NAA-SR-MEMO-5237, Atomics International, June 20, 1960
10. Koontz, R.L., et al., "Conventional Buildings for Reactor Containment," NAA-SR-10100, Atomics International, July 25, 1965

11. Koontz, R.L., et al., "Leakage Characteristics of Conventional Building Components for Reactor Housing Construction," Trans. Am. Nucl. 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

6.1.8 Topics for Future Standards Development

None

6.2 ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

Potential Category VI 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/postaccident 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 Category VI reactor facilities.

If airborne toxic agents are a hazard at a particular facility or experimental facilities introduce potential airborne hazards at a particular facility, an ESF ventilation system may be required to protect against the hazard. That facility should refer to the Design Guide for Category II reactors, Light and Heavy Water Reactors, for guidance on designing an ESF ventilation system.

CHAPTER 7

INSTRUMENTATION AND CONTROL

7.1 PROTECTION SYSTEM AND SAFETY CLASS 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 facility conditions in order to safely shut down the reactor and to mitigate the consequences of postulated accidents.

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 class instrumentation should provide the facility operator with sufficient information to readily determine the status of the protection system such that the reactor can be operated within the prescribed limits and initiate safety-related operator 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 device that performs the protective action). Also included are those signals that initiate the operation of engineered safety feature systems (see Chapter 6).

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, bypassed, 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 automatically or manually initiated protective actions must be taken in order to assure safe reactor

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

shutdown or to mitigate the consequences of postulated accidents. 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. Damage guidelines for a safety class reactor building (Section 6.1).
- D. Radiation and toxic gas exposure guidelines for facility operators (Section 6.2 and Chapters 12 and 15).
- E. Off-site radiological consequence guidelines (Chapter 12 and 15).

Those portions of the protection system and safety-related instrumentation that must function to limit postulated accident consequences within the above guidelines for damage and radiological consequences should be designated safety class protection and instrumentation systems. Electrical portions should be designed to meet Class IE requirements. Mechanical and fluid system components should be designed to a minimum of Safety Class 3.

Systems or portions of systems that are not required to limit accident consequences within the prescribed guidelines may be designated non-nuclear. Such systems may be designed to non-safety class instrumentation and control system guidelines of Section 7.2. Electrical components may be designated Non-Class IE and mechanical and fluid components designed to non-nuclear standards.

7.1.4.2 Guidelines for the Design of the Protection System

A. The protection system should be designed to detect abnormal facility conditions that require the initiation of protective actions (see Chapter 15) in order to safely shut down the reactor and to mitigate the potential consequences of accidents.

B. To the extent practical, the protection system inputs should be derived from signals that are direct measures of the desired facility parameters.

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.

F. The design of the protection system should incorporate suitable redundancy, and 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.

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

H. The protection system should be electrically separated (by means of buffer amplifiers or other suitable, qualified isolation devices) from non-safety class instrumentation and control systems which may receive protection system output signals 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. Isolation devices at the protection system boundary are considered to be part of the protection system.

I. The facility parameters monitored by the protection system, and the status of the protection system itself should be displayed on safety class 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 during normal, accident, and postaccident conditions.
3. The need for safety-related operator actions during accident and postaccident conditions.

J. 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 during any condition which may make the control room uninhabitable. As a minimum, this should include the ability to scram the reactor from a remote location.

K. 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

L. The protection system should be protected against severe natural and man-made phenomena (see Chapter 3).

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

N. Class IE power for the protection system may not be required where loss of all electric power places the reactor in a safe shutdown condition and 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.

O. The protection system components required to be operable should be qualified for all environmental conditions anticipated during normal operation and postulated accidents (see Section 3.11).

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. On-line tests should not prevent the protection system from performing its safety functions.

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 System

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. Class IE Electric Power System
- F. Fuel-Handling and Storage System
- G. Experimental Facilities
- H. Radiation Protection System

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 10 CFR 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 26, "Reactivity Control System Redundancy and Capability"
- J. GDC 29, "Protection Against Anticipated Operational Occurrences"
- K. GDC 64, "Monitoring Radioactivity Releases"
- L. 10 CFR 50.55a, "Code 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 N45.2.4), "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 of Nuclear Power Generating Stations"

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

6. IEEE 384-1974 (ANSI N41.14), "Criteria for Separation 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 Components 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. C 1-1T, "Instrumentation and Control Equipment Grounding and Shielding Practices"

2. C 15-2T, "Wide Range Neutron Flux Monitoring Channel"

3. C 15-6T, "Logarithmic Mean Square Voltage Intermediate Range Neutron Flux Monitoring System"

4. C 15-8T, "Direct Current Power Range Neutron Flux Monitoring System"

5. C 15-10T, "Logarithmic Count Rate Source Range Neutron Flux Monitoring System"

6. C 16-1T, "Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems"

7. C 16-2T, "Protection System Logic"

8. C 16-3T, "PPS Buffers"

9. C 16-4T, "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 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems"
4. USNRC Regulatory Guide 1.62, "Manual Initiation of Protective Action"
5. USNRC Regulatory Guide 1.75, "Physical Independence of Electrical Systems"
6. USNRC Regulatory Guide 1.105, "Instrument Spans and Setpoints"
7. 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 7.1, "Introduction (to I&C Chapter)," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 7.2, "Reactor Trip System," *ibid.*
3. USNRC Standard Review Plan 7.4, "Systems Required for Safe Shutdown," *ibid.*
4. USNRC Standard Review Plan 7.5, "Safety-Related Display Instrumentation," *ibid.*
5. USNRC Standard Review Plan 7.6, "All Other Instrumentation Required for Safety," *ibid.*
6. Branch Technical Position EICSB 10, "Electrical and Mechanical Equipment Seismic Qualification Program," *ibid.*

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

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

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

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

11. Hagan, E.W., "Reactor Protection Systems: Philosophies and Instrumentation Reviews from Nuclear Safety," ORNL-NSIC-11, Oak Ridge National Laboratory, July 1973

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

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

14. IAEA Safety Guide SG-D8, Draft, "Instrumentation and Control," *ibid.*

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

16. "Standards and Practices for Instrumentation," Fifth Edition, ISA, 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, 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"

5. ANSI N41.26, "Criteria for Design of Post-Accident Monitoring Systems"

7.1.8 Topics for Future Standards Development

None

7.2 NON-SAFETY CLASS INSTRUMENTATION AND CONTROL SYSTEMS

7.2.1 Scope

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

7.2.2 Safety Function

Non-safety class 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 class I&C system must not prevent the interfacing protection system from completing its safety function when required.

7.2.3 Definition

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

7.2.4 Design Guidelines

7.2.4.1 Guidelines for the Design of Non-Safety Class Instrumentation and Control Systems

- A. Non-safety class I&C systems may receive output information from the protection system.
- B. Non-safety class 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 class I&C system will not prevent a protection system from completing its safety function when required.
- E. Non-safety class I&C systems need not be qualified for the environmental conditions resulting from severe natural phenomena or postulated accidents.

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

7.2.4.2 Maintenance, Testing, and Inspection

A. Interconnections between a non-safety class I&C system and the protection system should be testable to verify electrical separation and independence of the two systems.

B. Non-safety class 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 class 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. Fuel-Handling and Storage System
- F. Fire Protection System
- G. Experimental Waste System
- H. Radioactive Waste System

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

7.2.5 Applicable 10 CFR 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 class instrumentation and control systems:

- A. GDC 13, "Instrumentation and Control"
- B. GDC 24, "Separation of Protection and Control Systems"
- C. GDC 64, "Monitoring Radioactivity Releases"

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 class instrumentation and control systems:

A. Industry Standards and Codes

1. ANSI C1-1971 (NFPA 70-1971), "National Electrical Code"
2. ANSI C2-1977, "National Electrical Safety Code"
3. IEEE 384-1974 (ANSI N41.14), "Criteria for Separation of Class IE Equipment and Circuits"
4. ISA "Standards and Practices for Instrumentation," 5th Edition
5. NEMA IS 1.1, "Enclosures for Industrial Controls"
6. NEMA ICS-1970, "NEMA Standards Publication, Industrial Controls and Systems"

B. RDT Standards

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

1. C 1-1T, "Instrumentation and Control Equipment Grounding and Shielding Practices"
2. C 17-4T, "General Instrumentation"
3. C 17-8T, "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. Hurrer, J.M., Beckerly, J.G., "Nuclear Power Reactor Instrumentation Systems Handbook" Volumes 1 and 2, TID-25952-P1 and -P2, USAEC, 1973 and 1974

B. Pending Industry Standards and Codes

1. ANS 4.3.4, Proposed, "Computer Application, Gamma-Digital Computers Used for Nuclear Power Generating Station Monitoring"

7.2.8 Topics for Future Standards Development

None

CHAPTER 8

ELECTRIC POWER

8.1 CLASS IE ELECTRIC POWER SYSTEMS

Category VI Reactors do not generally require a Class IE electric power system(s). This is justified by the following general characteristics of these 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 toxic materials or 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 VI 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 class portion of the facility electric-power system. It includes the off-site (preferred) power supply and non-safety class 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. Fuel-Handling and Storage System
- D. Fire Protection System
- E. Experimental Facilities
- F. Radioactive Waste 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 10 CFR 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-Class IE electric power systems:

- A. GDC 17, "Electric Power Systems"

B. GDC 24, "Separation of Protection and Control 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 C1-1971 (NFPA 70-1970), "National Electrical Code"
2. ANSI C2-1973, "National Electrical Safety Code"
3. IEEE 27-1969 (ANSI C37.20-1969), "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 FUEL-HANDLING AND STORAGE SYSTEM

9.1.1 Scope

Requirements for fuel-handling and storage systems are specified in ERDA Manual Chapter 0530, "Nuclear Criticality Safety." This section provides additional guidelines for the design of a fuel-handling and storage system. Figure 9.1-1 is a guide to the use of this section.

9.1.2 Safety Function

The safety function of the fuel-handling and storage system is to handle and store new and irradiated fuel in a manner which assures:

- A. No inadvertent criticality during handling or storage.
- B. No release of radioactivity during normal or accident conditions, beyond those allowed during appropriate operating conditions (see Chapter 15).

9.1.3 Definition

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

- A. Receiving fuel from a carrier and making necessary accountability inspections.
- B. New fuel storage prior to installation in the reactor.
- C. Handling fuel during movement between storage areas and the reactor, including core loading and unloading.
- D. Spent fuel storage for irradiated fuel and other radioactive components.
- E. Preparation of spent fuel and other related radioactive components for shipment to reprocessing or disposal areas.

9.1.4 Design Guidelines

9.1.4.1 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) their failure may result in potential off-site doses in

Figure 9.1-1. Guide to the use of section 9.1.

excess of 0.5 rem to the whole body, or the equivalent to any part of the body, and (2) they are required for maintaining new or spent fuel in a subcritical configuration.

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

9.1.4.2 Guidelines for the Design of Fuel-Handling and Storage Systems

A. Fuel-handling and storage systems should be designed to maintain fuel in a subcritical configuration (e.g., K_{eff} less than approximately 0.95) when fully loaded with fuel and flooded with water.

B. The handling and storage equipment should be designed to protect the fuel from the effects of severe natural and man-made phenomena (see Chapter 3).

C. Handling equipment and storage facilities should be adequately shielded to ensure that normal personnel exposures are maintained as low as reasonably achievable (see Chapter 12).

D. Radioactive gas concentrations released to the reactor building during core loading or unloading should be maintained at an acceptably low level. An ESF ventilation system (see Section 6.2) may be required to maintain acceptably low occupational exposures.

E. Lifting and transport equipment of the fuel-handling system should be designed to prevent damage to a fuel assembly.

F. Instrumentation should be provided to monitor parameters of the fuel-handling and storage system that are important to safety, and to alarm or initiate protective actions if required.

G. New or spent fuel to be transported from a 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."

H. Provisions should be made to ensure that the reactor core is maintained in a subcritical configuration during fuel handling when reactivity control assemblies may be removed to allow the fuel-handling system to have access to the fuel assemblies to be replaced or relocated in the core.

I. Mechanical portions of the fuel-handling and storage system should consider the design guidance in Section 3.9.

J. Safety class portions of the fuel-handling and storage system should be Seismic Category I (see Chapter 3).

9.1.4.3 Maintenance, Testing, and Inspection

Safety class components and equipment should be designed to facilitate periodic testing and inspection. This includes the fuel storage racks and any safety class plumbing system.

9.1.4.4 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. Protection System
- E. Non-Safety I&C System
- F. Non-Class IE Electric Power System
- G. Fire Protection System
- H. Radioactive Waste System
- I. 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.

9.1.5 Applicable 10 CFR 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 safety class and non-safety class 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"
- D. 10 CFR 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions"

9.1.6 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 Crane (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, "Validation of Calculational 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 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants"
2. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors"
3. USNRC Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety"
4. USNRC Regulatory Guide 3.43, "Nuclear Criticality Safety in the Storage of Fissile Materials"

9.1.7 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 9.1.1, "New Fuel Storage," NUREG 75/087, September 1975
2. Branch Technical Position APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants," (attached to SRP 9.1.4), *ibid*.
3. ANSI N16.11, "Evaluation of Neutron Interactions Between Units of Low ²³⁵U Enrichment"
4. ANSI N16.13, "Guide for Evaluating Interaction Between Units of Low Enriched Uranium Using the Surface Density Method"
5. ANSI N347, "Criteria for the Location, Design, Construction, Operations, and Maintenance of Facilities"

B. Pending Industry Standards and Codes

None

9.1.8 Topics for Future Standards Development

None

9.2 FIRE PROTECTION SYSTEM

9.2.1 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 a fire protection system that minimizes the adverse effects of fire on safety class structures, systems, and components. Figure 9.2-1 is a guide to the use of this section.

9.2.2 Safety Function

The safety functions of the fire protection system are to:

- A. Assure that the effects of a fire or a combustion explosion does not prevent other safety class components, systems, or structures from completing their safety functions when required.
- B. Minimize the potential for and mitigate the consequences of a graphite fire.

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

9.2.3 Definitions

In its broadest context, a fire protection system is a comprehensive program that includes all aspects of fire prevention, detection and 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 extinguishment in addition to administrative procedures, fire brigade organization, inspection, maintenance, training, quality assurance, and testing.

9.2.4 Design Guidelines

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

A. Portions of the fire protection system should be a minimum of Safety Class 3 if their failure may prevent safe reactor shutdown and cooling or result in potential off-site doses in excess of 0.5 rem to the whole body, or the equivalent to any part of the body.

B. Portions of the fire protection system passing through safety class structures should be designed to a safety class compatible with the structure.

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

9.2.4.2 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 and 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 both manual and automatic).

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

B. Fire protection system design, organization, and administration should consider the guidance in USNRC Regulatory Guide 1.120.

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

D. A single failure in the fire suppression system should not impair both the primary and backup fire suppression capability.

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

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

G. 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 sources (e.g., welding) is conducted under closely controlled conditions.
4. Work involving pyrophoric materials such as plutonium, sodium, and zirconium is conducted under closely controlled conditions.
5. Proper housekeeping requirements are followed that minimize accumulation of unwanted combustibles.

H. A reactor inert gas atmosphere or injection system should be considered to prevent oxidation of graphite.

9.2.4.3 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.

9.2.4.4 Interfaces With Other Systems

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

- A. Reactor building
- B. Core Assembly
- C. Protection System
- D. Non-Safety Instrumentation and Control System
- E. Non-Class IE Electric Power System
- F. Fuel-Handling and Storage System

G. 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.

9.2.5 Applicable 10 CFR 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"

9.2.6 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 a fire protection system:

A. Industry Standards and Codes

1. Fire Protection System 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 13-1973), "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 22-1974, "Water Tanks"
- q. NFPA 20-1974, "Standard for the Installation of Centrifugal Fire Pumps"
- 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"
- x. NFPA 214-1971, "Water Cooling Towers"
- 2. Fire Protection System Administration
 - a. NFPA 6-1974, "Industrial Fire Loss Prevention"
 - b. NFPA 7-1974, "Fire Emergencies Managements"
 - 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 E84-1971), "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. USNRC Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants"

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

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 for information only.

A. Background Information

1. USNRC Standard Review Plan 9.5.1, "Fire Protection System" 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 Brown's Ferry Fire," USNRC, February 1976

4. NFPA 49-1975, "Hazardous Chemicals Data"

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

6. 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"

9.2.8 Topics for Future Standards Development

None

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, thimbles, and test loops). The ERDAM 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), reactivity control system (see Section 4.2), the reactor building (see Section 6.1), or other safety class systems.

There may be no safety function associated with portions of the experimental facilities which do not interface with, or which are suitably isolated from, safety class systems.

10.3 DEFINITION

Experimental facilities include in-core irradiation facilities, in-core circulating loops for conducting fuel tests, 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. Experimental facility components interfacing with safety class components or structures should be designed to a safety class that is compatible with the interfacing components or structures.

B. In-core experimental facilities should be designed to a safety class that is compatible with the reactor core assembly to assure that a failure will not prevent safe shutdown of the reactor.

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

C. If the failure of an experimental facility does not prevent safe reactor shutdown, but may result in potential off-site doses in excess of the 10 CFR 100 guidelines, it should be designed to a minimum of Safety Class 2.

D. If the failure of an experimental facility does 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 guideline), it should be a minimum of Safety Class 3.

E. 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 or failure of experimental equipment. Potential reactivity effects should be within the range for which the reactivity control system (see Section 4.2) was designed.

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

C. The operation or failure of in-core experimental facilities should not cause damage to the reactor core assembly in excess of the guidelines in Section 4.1.

D. Experimental facilities should be designed to provide adequate protection against radiation from the following sources:

1. Streaming from beam tubes or other sources
2. Scattering from ex-core experiments
3. Irradiated sample handling
4. Activated portions of experimental facilities
5. Experimental facility cooling water
6. Other sources

Guidelines for occupational radiation exposure are included in the ERDA Manual, Chapters 0524 and 0525 (see Chapter 12).

E. Mechanical portions of experimental facilities should be designed with consideration to the guidelines of Section 3.9.

F. The instrumentation and controls associated with the reactor and experiments should include appropriate interlocks if operation of the experimental

facility can create conditions that may adversely affect the safety of the reactor. If the interlock circuits interface with the protection system, the guidelines in Section 7.1 should be considered in their design.

G. 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.

10.4.3 Maintenance, Testing, and Inspection

A. Experimental facility components that form part of the reactor core assembly should meet the maintenance, testing, and inspection requirements specified in Section 4.1.

B. Prior to loading in the reactor core, 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.

C. Interlock circuits between the reactor protection or control systems and the experimental facilities should be designed to facilitate periodic testing to verify proper operation of the interlock.

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

H. Radiation Protection 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 10 CFR GENERAL DESIGN CRITERIA

No General Design Criteria are specifically applicable to the design of experimental facilities. The GDCs that are applicable to interfacing facility systems should be considered in the design of experimental facilities that may potentially affect the safe operation of the 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 N399-1974, "Records and Reports for Research Reactors"

3. ANSI N401-1974, "Review of Experiments for Research Reactors"

4. ANSI N405-1975, "Safety Guide for the Performance of Critical Experiments"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 2.2, "Development 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. "Safe Operation of Critical Assemblies and Research Reactors, 1971 Edition," Safety Series No. 35, International Atomic Energy Agency, Vienna, 1971

2. "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. This chapter provides additional guidelines for the design of radioactive waste systems. Figure 11.1-1 is a guide to the use of this chapter.

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.

The ERDA Manual, Chapter 0513, "Effluent and Environmental Monitoring and Reporting," prescribes monitoring requirements that are applicable to systems used to monitor radioactive waste areas.

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."

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 Chapters 12 and 15).

11.3 DEFINITION

The radioactive waste system includes the equipment necessary to collect, store, sample, and treat radioactive waste materials and prepare them for reuse or disposal.

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 conditions (see Chapters 12 and 15).

Figure 11.1-1. Guide to the use of chapter 11.

11.4 DESIGN GUIDELINES

11.4.1 Guidelines for Determining the Minimum Safety Class of Radioactive Waste System Components

A. Mechanical portions of the radioactive waste system should be a minimum Safety Class 3 if their failure may result in potential off-site doses in excess of 0.5 rem. Other portions of the system may be designed to non-nuclear standards.

B. Portions of the radioactive waste system passing through safety class structures should be designed to a safety class compatible with the structure.

11.4.2 Guidelines for the Design of a Radioactive Waste System

A. In determining the capabilities and capacities required for a radioactive waste system, the following sources of potentially radioactive material should be considered:

1. Ventilation System (Filters)
2. Fuel-Handling and Storage System
3. Contaminated Components, or Equipment Used for Maintenance, Repair, Sampling or Testing
4. Cleaning Solvents and Rags
5. Reactor Building Sumps
6. 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 practicable.

11.4.3 Maintenance, Testing, and Inspection

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 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. 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 10 CFR 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 safety class or non-nuclear portions of radioactive waste systems:

- A. GDC 60, "Control of Releases of Radioactive Materials 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 N5.8-1967, "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 Effluent of Light Water-Cooled Nuclear 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"

CHAPTER 12

RADIATION PROTECTION

12.1 SCOPE

Requirements for radiation protection are specified in the following ERDA 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 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. Portions of the radiation protection system may be part of the protection system (see Section 7.1) and may automatically initiate protective actions or notify the operator of the need to take manual protective actions. 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.

B. The facility design should ensure adequate protection from direct and scattered radiation and airborne sources. One of the potential sources of radiation exposure particular to air cooled graphite reactors is from argon-41 which is created by the neutron activation of argon-40 cooling air. Methods of protection to be considered in the design of the facility include the following:

1. Permanent or temporary shielding.
2. Ventilation systems (see Section 6.2).
3. Equipment design to minimize surface contamination.
4. Area radiation zones and occupancy restrictions.
5. Fixed or portable monitoring and alarm systems.

6. Structures and equipment designed to minimize the release of argon-41 to the reactor building or other portions of the facility accessible during operation and to enhance the dispersion of exhausted reactor air coolant to maintain exposures ALARA.

C. Portions of the radiation protection system that initiate protective system actions and operation of engineered safety feature systems should consider the design guidelines in Section 7.1.

D. The radiation protection system may be required to monitor the following:

1. Reactor building radiation and airborne activity levels.

2. Radioactive waste system areas.
3. Fuel-handling and storage areas.
4. The facility environment, including gaseous and liquid effluent release paths.
5. Personnel with appropriate personal dosimeters to fulfill the administrative objective of the radiation protection system.
6. Contamination on personnel and surfaces.

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. Protection System
- C. Fuel-Handling and Storage System
- D. Experimental Facilities
- E. Radioactive Waste Systems

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

12.5 APPLICABLE 10 CFR 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 Reactor 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 Reading 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/ANS 6.4-1977 (N403), "Guidelines for Nuclear Analysis and Design of Concrete Shielding for Nuclear Power Plants"

m. ANSI N648-1975, "Specification for Aggregates for Radiation Shielding Concrete"

n. ANSI N649-1975, "Descriptive Nomenclature of Constituents of Aggregates for Radiation Shielding Concrete"

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 following RDT standard should be considered on a case basis:

1. E 9-1T, "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, "Measurement of Radionuclides in the Environment--Sampling and Analysis of Plutonium in Soil"
- e. USNRC Regulatory Guide 4.6, "Measuring 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, "Personal 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 Exposure 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 as Low as Reasonably Achievable," *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 Materials," October, 1976
15. "Safe Operation of Critical Assemblies and Research Reactors, 1971 Edition" IAEA Safety Series No. 35, International Atomic Energy Agency, Vienna, 1971
16. MIL-F-51068D, "Filter, Particulate, High Efficiency, Fire Resistant," 1974
17. MIL-STD-282, "Filter Units, Protective Clothing, Gas Mask Components and Related Products: Performance Test Methods," 1956
18. IAEA Safety Guide SG-05, Draft, "Operational Aspects of Radiological Protection," International Atomic Energy Agency, Vienna
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 Measurements of Continuous Area and Process Radiation Monitoring System for Light Water Reactors"

2. ANSI N221, Draft, "Specific Environmental Monitoring Program to Assess Operational Dose From LWR Power Plants"

3. ANSI N323, Draft, "Radiation Protection Instrumentation Test and Calibration"

4. ANSI N328, Draft, "Control of Radioactive Surface Contamination on Materials, Equipment and Facilities to be Released for Uncontrolled Use"

5. ANSI N336 (ANS 18.2.1), Draft, "Methods for Inferring Environmental Doses"

6. ANSI N346 (ANS 6.6), Draft, "Calculation and Measurement of Direct and Scattered Radiation from Nuclear Power Plants"

7. ANSI N404 (ANS 6.5), Proposed, "Glossary of Terms in Shielding and Dosimetry"

8. ANSI N410 (ANS 6.1.2), Proposed, "Neutron and Gamma Ray Cross Sections for Nuclear Radiation Protection Calculations"

9. ANSI N647 (ANS 15.12), Proposed, "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents"

10. ANSI N651 (ANS 6.7), Proposed, "Radiation Zoning for Design of Nuclear Power Plants"

11. ANSI N702 (ANS 18.1.2), Draft, "Radioactive Materials in Effluents from Light Water-Cooled Nuclear Power Plants"

12. ANSI N703 (ANS 18.1.3), Draft, "Monitoring of Radioactive Materials in Effluents from Light Water-Cooled Nuclear Power Plants"

13. ANSI N320, "Performance Specifications for Reactor Emergency Monitoring Instrumentation"

14. ANSI N317, "In-Plant Plutonium Monitoring for Personnel Protection"

15. ANS 58.4, Draft, "Criteria for the Radiological Technical Specifications for Nuclear Power Stations"

12.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

A. None

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 Non-Reactor 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 10 CFR 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 Plants"
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 18.17-1973, "Industrial Security for Nuclear Power Plants"
6. ANSI N399-1974, "Records and Reports for Research Reactors"
7. ANSI N401-1974, "Review of Experiments for Research Reactors"
8. ANSI N546-1976, "Medical Certification and Monitoring for Personnel Requiring Operator Licenses for Nuclear Power Plants"
9. ANSI/ANS 15.4-1977 (N380), "Selection and Training of Personnel for Research Reactors"

B. RDT Standards

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

1. F1-3T, "Preparation of Unusual Occurrence Reports"
2. F3-2T, "Calibration 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 Plans 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. NUREG-0094, "A Guide to the Licensing of Facility Operators, Including Senior Operators," USNRC, July 1976

13. WASH 1130, Revised, "Utility Staffing and Training for Nuclear Power," USAEC, June 1973

14. EPA-520/1-75-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents" U.S. Environmental Protection Agency, September 1975

B. Pending Industry Standards and Codes

1. ANSI N17.2 (ANS 15.16), Proposed, "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), Proposal, "Standard for Physical Security of Research Reactors"

6. ANS 15.18, Draft, "Standards for Administrative Controls for Research Reactors"

13.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

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 should be considered in the conduct of an initial test program.

14.2 APPLICABLE 10 CFR 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 standards, codes, and guides be considered in the conduct of the initial test program:

- A. Industry Standards and Codes

- 1. ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"

- 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 are distributed on only a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. IAEA Safety Guide SG-04, 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 Control 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. The chapter also recommends standards and guides to be considered in the analysis of postulated accidents at an air-cooled graphite 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 consequence 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 dose guidelines in Table 15.2-2 (25 rem total dose to the whole body, 300 rem total dose to the thyroid from iodine exposure, 150 rem to the bone and 75 rem to the lung from plutonium exposure, and 75 rem to any other organ).

Table 15.2-1

Spectrum of Postulated Accidents to be Considered in
the Design of an Air Cooled Graphite Reactor Facility

A. Reactivity and Power Distribution Anomalies

1. Uncontrolled positive reactivity insertion by the reactivity control system from a subcritical or low power startup condition (assuming the most unfavorable reactivity conditions of the core), including a reactivity control assembly or temporary control device removal error during refueling.
2. Uncontrolled positive reactivity insertion by the reactivity control system at the particular power level (assuming the most unfavorable reactivity conditions of the core) that yields the most severe results (low power to full power).
3. Reactivity control system maloperation (system malfunction or operator error).
4. Inadvertent loading and operation of a fuel assembly in an improper position.
5. Spectrum of accidents in which a reactivity control assembly is withdrawn from the reactor core.
6. Spectrum of accidents in which a reactivity control assembly inadvertently drops into the reactor core.
7. Spectrum of reactivity accidents due to operation or failure of experimental facilities.
8. Reactivity accident in which a moderating material may be introduced into the core.

B. Radioactive Release from A Subsystem or Component

1. Radioactive waste system leak or failure.
2. Fuel handling accidents.
3. Experimental facility or specimen failures.
4. Fires involving plutonium metals.
5. Contamination from bare plutonium.
6. Graphite fires due to the uncontrolled releases of stored energy.

Table 15.2-1 (Continued)

C. Anticipated Transients Without Scram

1. Inadvertent insertion of positive reactivity by the reactivity control system.
2. Loss of electric power.

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, shut-down, 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 and ≤ 300 rem thyroid

* 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

Summary of Facility Operating Conditions and the Corresponding
Frequency of Occurrence and Consequence Guidelines. (Continued)

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		

For those individuals who constitute the site emergency force, the guidance of ERDAM 0524 (particularly Part IV, "Guidance For Emergency Exposures During 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 response to a postulated design basis accident.

2. Off-Site Dose Guidelines

a. At the exclusion area boundary (see Subsection 2.1.1), 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 radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

b. At the outer boundary of the low population zone (see Subsection 2.1.1), an individual located at any point on the outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release resulting from the design basis accident (during the entire period of its passage) will not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

3. Dose Equivalence

Following some postulated accidents, the dose to the critical organ due to inhalation or ingestion of radionuclides may be more limiting than the whole body dose or the 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 equivalent 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 operation and/or failure do not exceed 0.5 rem to the whole body and if the system is not otherwise required to be a safety class system (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 operation and/or failure exceed 0.5 rem to the whole body and the system is not otherwise required to be Safety Class 1 or 2.

The final safety classification of a system is obtained by analysis of the design basis accidents. Because of specific features of an individual Category VI reactor facility, analysis of the design basis accidents may indicate that the consequences are not within the guidelines of Table 15.2-2. 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 of Table 15.2-2, or it may be necessary to increase the quality and reliability of the system by making the final safety class greater than the minimum safety class recommended by this Design Guide.

D. The spectrum of design basis accidents to be considered in the design of reactor facilities is listed in Table 15.2-1. Events such as floods, storms, earthquakes, and fires are not explicitly considered in this chapter. Severe natural phenomena (floods, storms, earthquakes) are discussed in Chapter 3 and fires are discussed in Section 9.2.

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 VI 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 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 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.

Figure 15.2-1. Approximate relationship between the facility operating conditions and the anticipated operational occurrences.

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.

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 which should be included in the accident analysis are:

1. The short-term diffusion estimates used for calculating the radiological consequences of postulated accidents should consider the guidance in USNRC Regulatory Guides 1.4, 1.24, and 1.25 regarding the acceptability of models and conservatisms appropriate to the design basis calculations.
2. The maximum steady-state reactor power level assumed in the accident analysis should be at a conservative margin above the actual maximum operating power level in order to account for instrument error and other factors.
3. 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.
4. 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 systems and to the analysis of the consequences of postulated accidents of air-cooled graphite reactor facilities, the General Design Criteria requires that systems such as the protection system, electric power system, and other safety class systems should be designed such that the system safety function can be accomplished with a single failure in the safety 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 system is to ensure that a single failure in a component or channel will not interfere with the proper operation or its redundant counterpart.

As described in previous chapters, Category VI reactors may have different system requirements than commercial power reactors. Category VI reactors may have fewer safety systems, and those which are provided may be functionally simpler than those in power reactors. 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 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, 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 of the active component such as a pressure boundary. A passive component, such as a pipe, tank, accumulator, valve body, 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, 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 includes 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 addition 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 VI 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 facility systems. Additional guidance on the application of the single-failure criterion is found in USNRC Regulatory Guide 1.53, ANSI N658, and IEEE 379.

Safety 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 a failure mode and effects analysis.

Undetectable failures differ from detectable failures in that they may exist in a safety 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 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 their failed mode. If this analysis indicates that the safety 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 purposes of applying the single-failure criterion, the period following a postulated accident may be 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 with the additional occurrence of (1) any single electrical failure, or (2) any active fluid system failure (e.g., failure of a valve to close).

Passive fluid system single failures are not considered during the short term because the high integrity of safety class fluid systems make it very improbable that a passive fluid system component will fail during the relatively brief short term.

The application of the single-failure criterion during the short term may establish minimum redundancy and capacity requirements for systems.

During the long term, assuming no prior application of the single-failure criterion during the short term, the single failure to be considered may be (1) any single electrical failure, (2) any active fluid system failure, or (3) a passive fluid system failure (e.g., pressure boundary failure).

The application of the single-failure criterion during the long term may establish minimum redundancy and capacity requirements for systems.

15.4 APPLICABLE 10 CFR 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 an air-cooled graphite reactor facility:

A. Industry Standards and Codes

1. ANSI N660 (ANS 15.4), Draft, "Criteria for Safety-Related Operator Actions"

2. IEEE 379-1972 (ANSI N41.2), "Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection 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 1.78, "Assumptions Used for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"

3. 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. USNRC Standard Review Plan 2.2.3, "Evaluation of Potential Accidents," NUREG 75/087, September 1975.
2. USNRC Standard Review Plan 2.3.4, "Short Term Diffusion Estimates," *ibid.*
3. USNRC Standard Review Plan 15.0, "Introduction (to Accident Analysis Chapter)," *ibid.*
4. USNRC Standard Review Plan 15.2.6, "Loss of Non-Emergency AC Power to the Station Auxiliaries," *ibid.*
5. USNRC Standard Review Plan 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition," *ibid.*
6. USNRC Standard Review Plan 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," *ibid.*
7. USNRC Standard Review Plan 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)," *ibid.*
8. USNRC Standard Review Plan 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," *ibid.*
9. USNRC Standard Review Plan 15.4.9, "Spectrum of Rod Drop Accidents," *ibid.*
10. USNRC Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," *ibid.*
11. USNRC Standard Review Plan 15.7.5, "Spent Fuel Cask Drop Accidents," *ibid.*
12. USNRC Standard Review Plan 15.8, "Anticipated Transients Without Scram," *ibid.*
13. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962
14. Slade, D.H. (Editor), "Meteorology and Atomic Energy - 1968," TID-24190, USAEC, 1968
15. Briggs, G.A., "Plume Rise," TID-25075, USAEC, 1969
16. Turner, D.B., "Workbook on Atmospheric Dispersion Estimates," AP-26, Revised, "USEPA Office of Air Programs, 1970
17. Streng, D.L., et al., "Review of Computational Models and Computer Codes for Environmental Dose Assessment of Radioactive Releases," BNWL-B-454, Battelle Northwest Laboratories, June 1976

18. 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 result in a more consistent approach to air-cooled graphite 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 should be considered when establishing technical specification limits for safe reactor facility operation.

16.2 APPLICABLE 10 CFR GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR be considered in establishing technical specifications for air-cooled graphite reactor facilities:

A. 10 CFR 50.36, "Technical Specification"

B. 10 CFR 50, Appendix I, Section IV, "Guide 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 air-cooled graphite reactor facilities:

A. Industry Standards and Codes

1. ANSI N378-1974, "Standard 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 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. USNRC Standard Review Plan 2.4.14, "Technical Specifications and Emergency Operation Requirements (Hydrology)," NUREG 75/087, September 1975

2. IAEA Safety Guide SG-03, Draft, "Safety Guide on Operational Limits and Conditions," International Atomic Energy Agency, Vienna

3. "Guide to Content of Technical Specifications for Nuclear Reactors," USAEC, November 1968

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."

The standards and guides that are listed in this chapter are recommended for consideration when establishing QA programs during the design, construction, and operational phases of a reactor facility.

Category VI reactor facilities are generally extremely low-risk facilities, their consequences of anticipated and postulated events being well within the guidelines of Chapter 15 for the various facility operating conditions. Therefore, many facility structures, systems, and components may be designated non-nuclear and designed to industry codes and standards. The quality of these non-nuclear systems and components may be more important for programmatic reasons than for safety. It is recommended that the listed standards and guides be appropriately considered for assuring quality control and maintenance of quality of non-nuclear systems and components.

Should a reactor facility require structures, components, or systems designed to the ASME Code (see Section 3.2), require a containment building (see Section 6.1), or Class IE Electric Power System (see Section 8.1), reference should be made to the Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors, for identification of additional quality assurance standards and guides that should be considered.

17.2 APPLICABLE 10 CFR GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR 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 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"
- d. ANSI N402-1976, "Quality Assurance Program for Research Reactors"
- 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"
- b. ANSI N45.2.13-1976, "QA Requirements for Control of Procurement of Items and Services of Nuclear Power Plants"

3. Construction Phase QA

- a. ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage and Handling Items for Nuclear Power Plants During the Construction Phase"
- b. ANSI N45.2.3-1973, "Housekeeping During the Construction Phase of Nuclear Power Plants"
- c. ANSI N45.2.4-1972 (IEEE 336), "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
- d. 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. F 1-2T, "Preparation of System Design Descriptions"
 - b. F 2-2T, "Quality Assurance Program Requirements"
2. Design Phase QA
 - a. F 2-9T, "Reliability Assurance"
3. Construction and Operation Phase QA
 - a. F 2-4T, "Quality Verification Program Requirements"
 - b. F 2-7T, "Acceptance Sampling Plans"
 - c. F 3-2T, "Calibration Program Requirements"
 - d. F 3-6T, "Non-destructive Examination"
 - e. F 3-10T, "Non-destructive Examination of Reactor Core

Components"

f. F 5-1T, "Cleaning and Cleanliness Requirements for Nuclear
Components"

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, Maintenance
of Nuclear Power Plant Quality Assurance Records"
 - c. USNRC Regulatory Guide 2.5, "Quality Assurance Program Require-
ments for Research Reactors"
2. Design Phase QA
 - a. USNRC Regulatory Guide 1.28, "Quality Assurance Program Require-
ments (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)"

b. USNRC Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"

c. USNRC Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants"

d. USNRC Regulatory Guide 1.123, "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

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

8. ANSI N45.2.22, Draft, "Supplementary Requirements for Inspection of Dimensional Characteristics"

9. ANSI N45.2.23, Draft, "Qualification of QA Program Audit Personnel for Nuclear Facilities"

17.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

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

9. IAEA Safety Guide SG-QA2, Draft, "Quality Assurance Record System," ibid.

10. IAEA Safety Guide SG-QA3, Draft, "Quality Assurance for Procurement of Items and Services for Nuclear Power Plant," ibid.

11. IAEA Safety Guide SG-QA5, Draft, "Management Control and Quality Assurance for Operation of Nuclear Power Plants," ibid.

12. IAEA Safety Guide SG-QA8, Draft, "Quality Assurance for Manufacture of Items for Nuclear Power Plants," ibid.

13. IAEA Safety Guide SG-QA10, Draft, "Quality Assurance Auditing for Nuclear Power Plants," ibid.

14. IAEA Safety Guide SG-02, Draft, "Inservice Inspection," ibid.

B. Pending Industry Standards and Codes

1. ANSI N45.2.12, Draft, "Requirements for Auditing of QA Programs for Nuclear Power Plants"

2. ANSI N45.2.14 (IEEE 467), Draft, "QA Program Requirements for the Design and Manufacture of Class IE Instrumentation and Electric Equipment for Nuclear Power Generating Stations"

3. ANSI N45.2.15, Draft, "Requirements for the Control of Hoisting, Rigging, and Transporting of Items at Nuclear Power Plant Sites"

4. ANSI N45.2.16 (IEEE P498), Draft, "Supplementary Requirements for the Calibration and Control of Measuring and Testing Equipment Used in the Construction and Maintenance of Nuclear Power Generating Stations"

5. ANSI N45.2.17, Draft, "QA Requirements for Control of Welding for Nuclear Power Plants"

6. ANSI N45.2.20, Draft, "Supplementary Quality Assurance Requirements for Sub-Surface Investigations Prior to Construction Phase of Nuclear Power Plants"

7. ANSI N45.2.21, Draft, "QA Requirements for Nuclear Power Plant Fuels"

CHAPTER 18

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