39/9
127
DR. 2142
MASTER

DESIGN GUIDE

FOR

CATEGORY I REACTORS

CRITICAL FACILITIES

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

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

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

DESIGN GUIDE FOR CATEGORY I REACTORS CRITICAL FACILITIES

W.J. Brynda and R.W. Powell

August 1978

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Prepared for the
Division of Operational and Environmental Safety

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.
UPTON, NEW YORK 11973

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy (DOE), nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

Price: Printed Copy \$9.25; Microfiche \$3.00

October 1978

265 copies

CONTENTS

CHAPTER		PAGES
	FOREWORD	vi
1	INTRODUCTION	
	1.1 Scope	1.1
	1.4 The Design Guide	1.2
	1.5 Definitions	1.7
2	SITE CHARACTERISTICS	
	2.1 Geography and Demography	2.1
	2.2 Meteorology	2.4
	2.3 Hydrology	2.7
	2.4 Geology and Seismology	2.10
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	3.7
	3.4 Water Level (Flood) Design	3.9
	3.5 Missile Protection	3.11
	3.6 Protection Against the Dynamic Effects Associated with Postulated Rupture of Piping	3.14
	3.7 Seismic Design	3.15
	3.8 Design of Category I Structures	3.19
	3.9 Mechanical Systems and Components	3.23
	3.10 Seismic Qualification of Category I Instrumentation and Electric Equipment	3.26

	3.11 Environmental Design of Mechanical and Electrical Equipment	3.28
	3.12 Welding of Nuclear Components	3.31
	3.13 Nuclear Reactor Facility Materials	3.33
	3.14 Decommissioning of Critical Facilities	3.35
4	REACTOR	
	4.1 Critical Assembly Core	4.1
	4.2 Reactivity Control System	4.5
5	FLUID FUEL SYSTEM	
6	ENGINEERED SAFETY FEATURES	
	6.1 Reactor Building	6.1
	6.2 Engineered Safety Feature Ventilation Systems	6.9
7	INSTRUMENTATION AND CONTROL	
	7.1 Protection System and Safety Class Instrumentation	7.1
	7.2 Non-Safety Class Instrumentation and Control System	7.9
8	ELECTRIC POWER	
	8.1 Class IE Electric Power Systems	8.1
	8.2 Non-Class IE Electric Power Systems	8.2
9	AUXILIARY SYSTEMS	
	9.1 Fuel Handling and Storage System	9.1
	9.2 Fire Protection System	9.7
10	EXPERIMENTAL FACILITIES	
11	RADIOACTIVE WASTE SYSTEMS	
12	RADIATION PROTECTION	
13	CONDUCT OF OPERATIONS	
14	INITIAL TEST PROGRAM	

- 15 ACCIDENT ANALYSIS
- 16 TECHNICAL SPECIFICATIONS
- 17 QUALITY ASSURANCE

FOREWORD

The Department of Energy (DOE) owns and operates a wide variety of nuclear reactors including critical facilities as part of a comprehensive program which supports the development of advanced energy systems. As required by the ERDA Manual, Chapter 0540, DOE-owned reactors are sited, designed, constructed, modified, operated, maintained and decommissioned 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).

Due to 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 lower operating power levels and fission product inventories, remote siting, 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, operating and siting characteristics and in their programmatic goals. A method has been developed for systematically and uniformly recommending codes, standards, guides, and general design criteria that apply to each facility in the diverse spectrum of DOE-owned reactors. The method is based on grouping the DOE-owned reactors into categories that are defined by generic reactor types. Within a given category, the DOE-owned reactors have generally similar design and 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 critical facilities.

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

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

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 I DESIGN GUIDE

Critical Facility Sub-Type	DOE Reactors Considered
Solid Fuel Systems	AFSR (Argonne Fast Source Reactor) (INEL) ATRC (INEL) (1) Big Ten (LASL) Comet (LASL) ETRC (INEL) (1) Flat Top (LASL) Honeycomb (LASL) Jezebel (LASL) Parka (LASL) Pool Critical Assembly (ORNL) (1) PNL Split Table (PNL-CML) Rocky Flats Vertical and Horizontal Split Tables (RFP-NFS) ZPR 6 (ANL) ZPR 9 (ANL) ZPPR (ANL) (INEL)
Liquid Fuel Systems	Kinglet (LASL) PNL Plutonium Solution System (CML) Rocky Flats Solution System (RFP-NFS) Rocky Flats Tank Reservoir (RFP-NFS)
Gaseous Fuel Systems	Plasma Core Assembly (LASL)

Notes: (1) This Design Guide is applicable when these facilities perform experiments and operate within the general characteristics described in Section 1.3.1 of this guide. Refer to Design Guide for Category III, Pool Type Reactors.

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 also generally applicable to the design of critical facilities. The Design Guide includes design guidelines which provide interpretations of the standards, codes, and guides in order to facilitate their application to critical facilities. When existing criteria do not provide adequate design guidance, the design 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 a user to note and transmit comments or suggestions that he may generate in the use of 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

1.1 SCOPE

The Department of Energy (DOE), 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.

1.2 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 critical facilities 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.

1.3 CATEGORY I REACTORS

This guide deals principally with the design and functional requirements of Category I critical facility structures, components, and systems.

1.3.1 General Category I Reactor Characteristics

Critical facilities consist of those systems, structures and components that are necessary to safely perform experiments with fissionable material which may be at or near nuclear criticality. The principle purpose for performing critical experiments is to study the neutron behavior within a critical assembly. A critical facility may contain one or more critical assemblies.

A critical assembly is the neutron-multiplying device or system which is operated at or near nuclear criticality for the performance of critical experiments, and may be assembled from various fissile, moderator, and reflector materials.

The critical assembly normally operates at near zero or low power level (a few tens of watts or less) such that the energy produced by fission or decay heat is insufficient to require auxiliary cooling. As such, critical assemblies do not require decay heat removal capability, or the incorporation of emergency core cooling systems or ultimate heat sinks as part of the critical facility.

1.3.2 Category I Reactor Subgroups

The critical assemblies are subdivided into two broad subgroups, solid fueled and fluid fueled critical assemblies.

The solid fueled critical assemblies consist of:

1. Systems or devices commonly referred to as assembly machines.
2. Fuel assemblies for large reactor cores to study the neutron behavior of future large reactor cores.

The fluid fueled critical assemblies consist of:

1. Liquid fuel systems (e.g., UO_2SO_4 solution systems)
2. Gaseous fuel systems (e.g., UF_6)

A critical facility may include one or more of these critical assembly subgroups. When more than one critical assembly is housed in the same reactor building only one critical assembly is normally operated at any one time.

1.4 THE DESIGN GUIDE

1.4.1 Design Guide General Safety Philosophy

The design of DOE-owned reactors is based on fundamental safety philosophies that are similar 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. Due to significant differences between Category I critical facilities and large research and power reactors, it is possible to simplify the design of the critical facility while still adequately protecting health and safety. In general, critical facilities present less risk to health and safety. The critical assemblies are operated intermittently and at extremely low power levels, creating minimal fission product inventories in comparison to large reactors which are operated continuously. Remote siting in combination with the minimal fission products inventory may result in negligible off-site consequences from postulated accidents for many critical facilities. However, consideration must be given to materials used by the critical facilities (e.g., plutonium) which can present significant hazards to health and safety. Critical facilities do not require engineered safety features (ESF) systems (e.g., ECCS, ultimate heat sinks) to protect the critical assembly core. As a result, Category I critical facilities require fewer physical barriers to the release of radioactivity to the environment than large research or power reactors in order to remain within the applicable effluent concentration, off-site, and on-site dose guidelines specified in ERDA Manual, Chapter 0524, and in 10 CFR 100. For example, some critical facilities may utilize a reactor building that does not control the release of radioactivity to the environment like a containment. Defense-in-depth is provided by the physical isolation of the reactor building from other on-site buildings and the site boundary. In addition, inherent safety features such as reduced fission products, minimal reliance on engineered safety feature systems to mitigate accidents provide the required reasonable assurance that the DOE-owned critical facility can be operated without undue risk to health and safety.

Table 1.4-1 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 1.4-1 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 1.4-1 guidelines, the radiological exposures are maintained as low as reasonably 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

Critical facility structures, components, and systems that protect health and safety (radiological dose limits within Table 1.4-1) are those required 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 varied designs of DOE-owned critical facilities, it is not within the scope of this Design Guide to identify only those safety functions that may be applicable to a specific facility. The safety functions in this guide are the customary performance characteristics that have generally been accepted by industry experience.

The implementation and maintenance of the safety functions is achieved through the design of components, systems and structures to quality standards that are commensurate with the importance of their safety function(s). A graded set of quality standards establish the basis for defining the design requirements that apply to each of three nuclear safety classes (Safety Class 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 design guidelines that are applicable to a specific critical 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

TABLE 1.4-1. SUMMARY OF GENERAL SAFETY PHILOSOPHY FOR
FACILITY OPERATION IN TERMS OF THE CORRESPONDING
FREQUENCY OF OCCURRENCE AND 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, ASSEMBLY OF CORE	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 EXCURSION	SMALL FRAC- TION 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 RADIO- ACTIVE MATERIAL AND BECAUSE THEY REPRESENT UPPER BOUNDS ON FAILURES OR ACCIDENTS WITH A PROBABILITY OF OCCURRENCE SUFFICIENTLY HIGH TO REQUIRE CONSIDERATION IN DESIGN. GENER- ALLY, A POSTULATED ACCIDENT IS NOT CONSIDERED IN DESIGN IF THE PROBABILITY OF ITS OCCUR- RENCE RESULTING IN UNACCEPTABLE CONSEQUENCES IS LESS THAN 10^{-7} PER YEAR	NUCLEAR BURST	$\leq 10CFR100$ AND FROM PLUTONIUM: ≤ 75 rem LUNG ≤ 150 rem BONT	< 25 rem WHOLE BODY AND ≤ 300 rem THYROID AND FROM PLUTONIUM: ≤ 75 rem LUNG ≤ 150 rem BONE

* AS LOW AS REASONABLY ACHIEVABLE.

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, 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-safety class, non-nuclear (see Chapter 3).

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 limit consequences of Table 1.4-1 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 only applicable to a particular facility. The Design Guide provides 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 1.4-1) or for programmatic reasons self-imposed at the particular site.

1.4.2 Use of the Design Guide

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

Many existing industry standards and codes and USNRC Regulatory Guides, although written primarily for commercial power reactors, may also be applicable to the design of critical facilities. Where necessary, design guidelines provide interpretations of the standards, codes, and guides in order to facilitate

their application to critical facilities. 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.

1.4.3 Design Guide Organization

Figure 1.4-1 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 critical facility or modifying an existing critical facility. As illustrated in Figure 1.4-1, 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 1.4-2 lists the systems that are typically associated with each subgroup of Category I critical facilities and also references the Design Guide chapter or section that provides the detailed system design information. A comparable listing for each subgroup of Category I critical facilities is also found in Table 3.2-2, which also summarizes the principal design codes, safety class, quality group, and seismic category generally applicable to the design of typical Category I critical facility systems. Due to the significant differences in programmatic roles, siting, and the type and form of fissile material used among Category I critical facilities, it is anticipated that the features of an individual facility may vary substantially from the listings in Table 3.1-2 while still adequately protecting health and safety.

The detailed design guidelines recommended for each system or structure is presented in an individual chapter or section of this Design Guide (Chapters 4-12). Each chapter or section describing a system includes the following information:

- A. Description of the System Safety Function.
- B. Definition of the System.

Figure 1.4-1. Design Guide Organization.

Table 1.4-2. Typical Systems for Category I Critical Facilities

Systems ⁽¹⁾	Applicable Design Guide Section	
	Reactors Using Uranium	Reactors Using Plutonium
Reactor Core Assembly	4.1	4.1
Reactivity Control System	4.2	4.2
Reactor Fluid System	5 ⁽²⁾	5 ⁽²⁾
Reactor Building	6.1 ⁽³⁾	6.1 ⁽⁴⁾
ESF Ventilation System	---	6.2
Protection System	7.1 ⁽⁵⁾	7.1 ⁽⁶⁾
Non-Safety I & C System	7.2	7.2
Class IE Electric System	8.1 ⁽⁷⁾	8.1 ⁽⁸⁾
Non-Class IE Electric System	8.2	8.2
Fuel Handling & Storage System	9.1	9.1
Fire Protection System	9.2	9.2
Experimental Facilities	10	10
Radioactive Waste System	11	11
Radiation Protection System	12	12

NOTES:

- (1) This systems listing should not be interpreted as a mandatory minimum systems requirement, but rather as a listing of systems that would be found at a hypothetical "average facility" and their corresponding Design Guide section or chapter references.
- (2) For liquid and gaseous fueled critical assemblies.

Table 1.4-2 (Cont'd) Typical Systems For Category I Critical Facilities

- (3) Generally, the reactor building need not be capable of controlling the release of radioactivity to the environment.
- (4) Generally requires a reactor building capable of controlling the release of radioactivity to the environment.
- (5) Generally for critical facility operating interlocks and critical assembly trip functions only.
- (6) For critical facility operating interlocks, critical assembly trip, and engineered safety feature systems used to limit the uncontrolled release of radioactivity.
- (7) As necessary to support protection system operation.
- (8) As necessary to support protection system and engineered safety feature systems.

C. Diagrams which 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.

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, this section provides interpretations of the requirements of existing codes, standards, and guides in order to facilitate their application to critical facilities. 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, and administration 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).

1.4.4 Maintenance of the Design Guide

The continuing development and revision of standards, codes, and guides that are applicable to Category I critical facilities will necessitate the periodic updating of the Design Guide in order to ensure that the guidance being provided to the Design Guide users 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 periodically revising the Design Guides and for providing additional information as necessary to the Design Guide users during the period between revisions.

1.5 DEFINITIONS

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

1. Class IE

The safety classification assigned to that electric equipment and those 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).

2. 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).

3. Critical Assembly

A device or physical system for performing critical experiments. The energy produced by fission in a critical assembly is insufficient to require auxiliary cooling, and the power history is such that a significant inventory of long-lived fission products is not accumulated (ANSI N405-1975).

4. Critical Experiment

An experiment or series of experiments performed with fissionable material which may be at or near nuclear criticality. The principle purpose is the study of neutron behavior within a critical assembly (ANSI N405-1975).

5. Critical Facility

That complex of structures, systems, and components that are used to prepare, operate, or support the operation of a critical assembly. These may consist of but are not limited to: reactor building; laboratories, offices and control rooms; fissile material storage areas including storage tanks, transfer systems, and solution preparation areas; maintenance and service areas.

6. Design Basis Accidents

Those limiting accidents of low probability that are postulated and analyzed to establish design requirements of all safety-class equipment (ANSI 213)

7. 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).

8. Off-Site Dose

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

9. On-Site Dose

Potential radiological exposure to persons within the exclusion area boundary.

10. Operating Basis Earthquake (OBE)

That earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material that could reasonably be expected to affect the facility site during the operating life of the facility. It is that earthquake which produces the vibratory ground motion for which those features of the critical facility necessary for continued operation without undue risk to health and safety are designed to remain functional (10 CFR 100, Appendix A).

11. Operating Condition

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

12. Reactor Building

A structure that's primary function is to house the critical assembly and associated experimental systems during the performance of critical experiments. It may also house a fissile material storage area and auxiliary equipment to support operations.

13. Safe Shutdown Earthquake (SSE)

That earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to assure:

(a) The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 100 (10 CFR 100, Appendix A).

14. Safety Class

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

15. Safety Function

Performance characteristics required to assure safe shutdown of a critical assembly and/or for the prevention of an uncontrolled release of radioactivity to the environment from the critical facility.

16. 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 which guard against the uncontrolled release of radioactivity. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency (10 CFR 50.36c, Revised).

17. Seismic Category I

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

18. Severe Natural Phenomena

Those conditions which are postulated as the most severe that can reasonably be derived from the history and properties of the site and surroundings (ANSI N212).

19. 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 ERDA 0524).

CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Guidelines for Critical Facility Siting

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

B. Highways, railways, and waterways which traverse the site should be sufficiently distant from critical facility structures so that routine use of these routes is not likely to interfere with normal critical facility operation. Hazardous material shipments over these routes should be evaluated for potential impact on the critical facility.

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

D. The 10 CFR 100 guidance concerning the "Low Population Zone" surrounding a facility and the "Population Center Distance" should be considered in siting a critical facility. Maximum accident dose guidelines for persons at the outer boundary of the "Low Population Zone" are defined in Chapter 15. In the event 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 Distances" 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 critical 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 critical 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 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites"
2. USNRC Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations"
3. USNRC Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations"
4. USNRC Regulatory Guide 4.11, "Terrestrial Environmental Studies for Nuclear Power Stations"

2.1.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or are distributed only on 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 (ANS 2.6), Proposed, "Guidelines for Determining Present and Future Population Distribution at Power Reactor Sites"

2. ANSI N635 (ANS 2.12), Draft, "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 Critical Facility Siting

A. The critical 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 exists, an onsite meteorological program should be established using suitably accurate and reliable meteorological sensors and a suitable system for data recovery. A single site meteorology program, which also provides data for other facilities on site, would adequately provide this monitoring capability.

C. The short-term and long-term diffusion estimates used for calculating the radiological consequences of postulated accidents and normal releases respectively should consider the guidance in TID-24190 regarding the acceptability of models and conservatisms appropriate to the design basis calculations.

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 critical 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 critical 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 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from LWRs"

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

5. USNRC Regulatory Guide 3.35, "Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Plutonium Processing and Fuel Fabrication Plant"

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

A. Background Information

The Standard Review Plans listed below provides 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. Holzworth, G. C., "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Office of Air Programs, USEPA, January 1972

11. Slade, D. H., (Ed.), "Meteorology and Atomic Energy - 1968," TID-24190, Division of Technical Information, USAEC, 1968

12. Yanskey, G. R., Markee, E. H., and Richter, A. P., "Climatology of the National Reactor Testing Station," IDO-12048, Idaho Operations Office, USAEC, 1966

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

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

B. Pending Industry Standards and Codes

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

2.2.5 Topics for Future Standards Development

None

2.3 HYDROLOGY

2.3.1 Guidelines for Critical Facility Siting

A. The potential for flooding from each of the following sources and events should be considered in evaluating the suitability of a critical 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 site.
6. Ice loadings from water bodies.

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

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 critical 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 critical facility site.

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 15.7-1977 (N379), "Research Reactor Site Evaluation"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"
2. USNRC Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants"
3. USNRC Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I"
4. USNRC Regulatory Guide 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 are distributed only on 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 2.4.10, "Flood Protection Requirements," *ibid.*
8. USNRC Standard Review Plan 2.4.12, "Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters," *ibid.*

9. USNRC Standard Review Plan 2.4.14, "Technical Specifications and Emergency Operation Requirements," ibid.

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

11. Reports of the Corps of Engineers, U.S. Army (These reports address many specific hydrology related topics.)

12. Hydrometeorological Reports and Technical Papers of the U.S. Weather Bureau (now U.S. Weather Service, NOAA), Hydrometeorological Branch (These reports address many specific hydrology related topics.)

B. Pending Industry Standards and Codes

1. ANSI N515 (ANS 2.13), 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 Critical Facility Siting

A. The following geologic and seismic features should be considered in evaluating the suitability of a critical 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 critical 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 critical 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 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 are distributed only on a limited basis as preliminary drafts are listed for information only.

A. Background Information

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

1. USNRC Standard Review Plan 2.5.1, "Basic Geologic and Seismic Information," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 2.5.2, "Vibratory Ground Motion," *ibid.*
3. USNRC Standard Review Plan 2.5.3, "Surface Faulting," *ibid.*
4. USNRC Standard Review Plan 2.5.4, "Stability of Subsurface Materials and Foundations," *ibid.*
5. USNRC Standard Review Plan 2.5.5, "Stability of Slopes," *ibid.*
6. Cohee, G. V., 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, Draft, "Earthquakes and Associated Topics for Nuclear Power Plants," International Atomic Energy Agency, Vienna
11. IAEA Safety Guide SG-S2, Draft, "Seismic Analysis and Testing of Nuclear Power Plants," *ibid.*

B. Pending Industry Standards and Codes

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

2. ANSI N174 (ANS 2.11), Proposed, "Guidelines for Evaluating Site-Related Geotechnical Parameters at Power Reactor Sites"

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

2.4.5 Topics for Future Standards Development

None

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 INTRODUCTION TO DESIGN CLASSIFICATION

Critical facility structures, components, and systems required for preventing the uncontrolled release of radioactivity to the environment 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. Figure 3.1-1 is a guide to the use of Chapter 3. 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.

10 CFR 50, Appendix A, General Design Criterion 2 requires that safety class structures, systems, and components at a nuclear power plant be designed to withstand the effects of severe natural phenomena such as earthquakes, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions. As illustrated in Figure 3.1-1, the severe natural and man-made phenomena which should be considered in the minimum design of facility safety class systems are those which may result in a release of radioactivity to the environment with radiological consequences in excess of the guidelines in Chapter 15. Probability guidelines applicable to the determination of the spectrum of severe natural phenomena to be considered in design are also described in Chapter 15. Some of these severe natural phenomena, such as hurricanes and floods, may be forecast with sufficient accuracy and timeliness to allow some protective actions to be taken in advance of their most severe effects on a critical facility. Other severe phenomena, such as earthquakes, 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, critical facilities may be operated on a more flexible schedule that permits shutting down the critical assembly when the facility is threatened by impending severe natural phenomena. At these facilities, some safety class systems required during critical assembly operation may not require protection from some severe natural phenomena if a timely shutdown of the critical facility 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 critical assembly in a safe shutdown condition and for preventing an uncontrolled release of radioactivity to the environment should be protected against the effects of the severe natural phenomena.

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.

At critical facilities, Technical Specification limits on facility operation during 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 critical 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. Critical facility safety 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.

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

Critical facility structures, components, and systems specifically designed to withstand, without loss of safety function, the effect of the Safe Shutdown Earthquake (SSE) are designated Seismic Category I. The SSE is based on an evaluation of the maximum earthquake potential of the critical 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. Structures, components, and systems required to prevent radiological release consequences during a safe shutdown earthquake from exceeding the guideline values in Chapter 15 for the associated frequency of occurrence should be Seismic Category I. Considering the operational lifetime of critical facility structures, components, and systems, and the resulting risk from seismic events, many critical assemblies may not be required to be Seismic Category I. Whereas the long lived reactor building and any required ESF systems (i.e., reactor building exhaust system) may be designated Seismic Category I).

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.

Critical facilities do not generally require Class IE electric power systems due to their inherent fail safe design on loss of electric power. Portions of the protection system may need to be designed to Class IE standards, as described in Section 7.1.

3.2.3 Determination of System Safety Class

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 (Non-Nuclear) corresponds with quality group D. USNRC Regulatory Guide 1.26 requires that a system be designed to a safety class if the off-site doses due to failure of the system can exceed 0.5 rem to the whole

body. If the off-site doses due to a failure of the system do not exceed 0.5 rem, the system may be designed to non-nuclear standards. Table 3.2-1 presents general guidelines for determining the minimum safety class of Category I critical 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 I critical facilities. Table 3.2-2 is not to be interpreted as a listing of required systems for each subgroup of Category I critical facilities. It is rather a listing of systems that would be found at a hypothetical "average facility." Due to the significant differences in programmatic roles, siting, and the types and forms of fuel among Category I critical facilities, 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 this Design Guide. If any accidents would result in doses exceeding the guidelines in Chapter 15, some system changes may be needed. Alternatively, it may be necessary to design certain systems to a final safety class that is higher than the minimum safety class in order to increase the quality and reliability of the equipment.

As described below, determining the safety class of structures, components, and systems provides the designer with the information necessary for defining the codes, standards and guides that should be considered in the design of critical facility equipment.

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

Critical facilities generally do not require structures, components and systems designed to the ASME Code. In the event a critical facility structure, system or component must be designed to the ASME Code guidance for the application of the ASME Code can be found in the Design Guide for Category II Reactors, Light and Heavy Water Cooled Reactors.

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 they may be 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 to the extent practical reflect consistency with similar requirements for equipment in the same safety class that is designed to the ASME Code and ensure the safe shutdown of the critical assembly and/or to prevent an uncontrolled release of radioactivity with radiological consequences in excess of the guidelines in Chapter 15.

TABLE 3.2-1
GENERAL SAFETY CLASS DEFINITIONS⁽¹⁾

Safety Class 1 (Quality Group A)

Critical facilities do not contain 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, containment isolation and containment cleanup systems.)

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 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 I REACTORS
SOLID AND FLUID PLUTONIUM FUEL SYSTEM

SYSTEM OF COMPONENT	DESIGN GUIDE SECTION	MINIMUM SAFETY CLASS	QUALITY GROUP	SEISMIC CATEGORY	PRINCIPAL DESIGN CODE (h) OR STANDARD
REACTOR CORE ASSEMBLY	4.1	-	-	-	-
REACTIVITY CONTROL SYSTEM	4.2	-	-	-	-
REACTOR FLUID SYSTEM	5	NN	D	-	-
REACTOR BUILDING	6.1	2 or 3 (a)	B or C	I	A58.1-1972, A89.1-1972, AISC
ESF VENTILATION SYSTEM	6.2	3	C	I	-
PROTECTION SYSTEM	7.1	(b)	-	I	IEEE 279
NON-SAFETY INSTRUMENTATION AND CONTROL SYSTEM	7.2	-	-	-	NEMA ICS-1970
NON-IE ELECTRIC POWER SYSTEM	8.2	-	-	-	ANSI C-1971, NEMA MG 1-1972
FUEL HANDLING AND STORAGE SYSTEM	9.1	3	C	I	-
FIRE PROTECTION SYSTEM	9.2	NN	D	-	NFPA 802-1974
EXPERIMENTAL FACILITIES	10	-	-	-	-
RADIOACTIVE WASTE SYSTEM	11	NN	D	-	-
RADIATION PROTECTION SYSTEM	12	3 (d)	C	-	-

TABLE 3.2-2 (CONTINUED)
TYPICAL EQUIPMENT CLASSIFICATION FOR CATEGORY I REACTORS
SOLID AND FLUID URANIUM FUEL SYSTEM

SYSTEM OR COMPONENT	DESIGN GUIDE SECTION	MINIMUM SAFETY CLASS	QUALITY GROUP	SEISMIC CATEGORY	PRINCIPAL DESIGN CODE (h) OR STANDARD
REACTOR CORE ASSEMBLY	4.1	-	-	-	-
REACTIVITY CONTROL SYSTEM	4.2	-	-	-	-
REACTOR FLUID SYSTEM	5	NN	D	-	-
REACTOR BUILDING	6.1	3 (e)	C	I	A58.1-1972, A89.1-1972, AISC
PROTECTION SYSTEM	7.1	(f)	-	-	IEEE 279
NON-SAFETY INSTRUMENTATION AND CONTROL	7.2	NN	D	-	NEMA ICS-1970
NON-IE ELECTRIC SYSTEM	8.2	NN	D	-	ANSI C1-1971 NEMA MG 1-1972
FUEL HANDLING AND STORAGE SYSTEM	9.1	3	C	I	-
FIRE PROTECTION SYSTEM	9.2	NN	D	-	NFPA 802-1974
EXPERIMENTAL FACILITIES	10	-	-	-	-
RADIATION PROTECTION	12	3 (d)	C	-	-

TABLE 3.2-2 (CONTINUED)

TYPICAL EQUIPMENT CLASSIFICATION FOR CATEGORY I REACTORS

Notes:

- a. Generally requires a reactor building capable of controlling the release of radioactivity to the environment (see Guidelines Section 6.1).
- b. For critical facility operating interlocks, critical assembly trip, and engineered safety features used to limit the uncontrolled release of radioactivity.
- c. To support protection system and engineered safety features.
- d. Portions of the system used to initiate protective actions. The remaining may be NN (see Section 12.3).
- e. Generally the reactor building need only protect the critical assembly(s) from natural and man-made phenomena.
- f. For critical facility operating interlocks, and critical assembly trip.
- g. To support protection system operation.
- h. Safety class systems are subject to the quality assurance requirements of 10 CFR 50, Appendix B. Non-Nuclear systems are subject to normal industrial QA practices.

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. Safety Class 1, 2, and 3 systems 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 N182, "Physical Protection for Systems and Components Important to Safety"

B. RDT Standards

None

C. Guides

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

3.2.6 Supplementary Information

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

Safety class systems identified in Section 3.2 should consider the following wind loading design guidelines.

3.3.1 Design Guidelines

A. Structures, components and systems should be designed to withstand or be protected from the effects of wind and tornado loadings if they must function during or after such natural phenomena to maintain radiological releases within the guideline values of Chapter 15. As a minimum, the critical assembly 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 tornadoes if the critical assembly 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 loadings resulting from the design basis tornado is likely to exceed any other wind loading. Nevertheless, there may be some structures, for example at coastal locations, where a separate wind loading should be considered. ANSI A58.1-1972 and ASCE Paper 3269 describe two methods of determining the loading resulting from the operating basis wind speed.

3.3.2 Applicable 10 CFR General Design Criterion

It is recommended that the following 10 CFR 50, Appendix A, General Design Criterion be considered in designing a critical 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 critical 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 are distributed only on 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)

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"

3.3.5 Topics for Future Standards Development

None

3.4 WATER LEVEL (FLOOD) DESIGN

Safety class systems identified in Section 3.2 should consider the following flood protection design guidelines.

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 maintain radiological releases to within the guideline values of Chapter 15. As a minimum, the critical assembly core and the fuel in storage should be protected against flooding in order to prevent an inadvertent criticality. Protection requirements for other facility systems should be based on the above general guidelines.

B. Other systems need not be protected against the effects of flooding. A critical facility, thus, need not be designed to remain fully operational during a flood. Technical Specification limits on facility operation may provide a satisfactory alternative by requiring that the reactor be shut down in advance of forecast flooding.

C. The magnitude of the design basis flood, including attendant wind-generated wave activity, for which the critical facility is designed should be determined based on an assessment of the flood risk from all possible causes, including seismically induced floods and dam failures.

D. Methods of flood protection to be considered should include

a. Dry Site

b. Exterior Barriers

(1) Levee

(2) Seawall or Floodwall

(3) Bulkhead

(4) Revetment

(5) Breakwater

3.4.2 Applicable 10 CFR General Design Criterion

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

A. Industry Standards and Codes

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

2. ANSI/ANS 58.3-1977 (182), "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"

3.4.4 Supplementary Information

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

A. Background Information

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

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

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

B. Pending Industry Standards and Codes

None

3.4.5 Topics for Future Standards Development

None

3.5 MISSILE PROTECTION

Structures, components and systems should be protected from the effects of potential missiles if as a result of such missiles they are required to maintain radiological releases to within the guideline values of Chapter 15. If failure of the structures, components or systems results from the effects of missiles, but the radiological release consequences do not exceed the guideline values of Chapter 15 for the associated frequency of occurrence, then these structures, components and systems need not be protected from such missiles. Missiles originating within the critical facility are usually generated by equipment failures or accidents at the critical facility. Due to the low operating temperature and pressure of critical experiments, the potential hazard from missiles generated within the critical assembly is minimal and protection is generally not required. Other facility equipment and systems should be examined, however, to identify any other potential sources of missiles. Externally generated missiles may be caused by naturally occurring phenomena such as tornados and earthquakes or by accidents not related to the operation of the facility such as aircraft crashes or explosions on nearby transportation routes.

3.5.1 Design Guidelines

3.5.1.1 Missiles Generated at the Critical Facility

A. Among the facility-generated missiles to be considered as potential hazards are those arising from failure of experimental facilities, high-speed 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 likelihood for generating missiles should be located such that safety class systems will not be damaged by potential missiles. 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 critical assembly and other experimental system components.

3.5.1.2 Missiles Generated by Natural Phenomena

A. Tornados, 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.

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

B. The peak overpressure resulting from an explosion may be equated to the dynamic wind pressure from the design basis tornado.

C. In most cases, designing a facility to withstand the design basis tornado will also provide adequate protection 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.

D. The assessment of aircraft hazards at the size 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 critical 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 critical 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 are distributed only on a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," *ibid*.

2. USNRC Standard Review Plan 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," *ibid*.

3. USNRC Standard Review Plan 3.5.1.6, "Aircraft Hazards," *ibid*.

4. USNRC Standard Review Plan 3.5.3, "Barrier Design Procedures," *ibid*.

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

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

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

B. Pending Industry Standards and Codes

None

3.5.5 Topics for Future Standards Development

None

THIS PAGE IS INTENTIONALLY LEFT BLANK

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

Category I reactor facilities do not have high energy piping systems. Protection against the dynamic effects of piping rupture is therefore not required.

3.7 SEISMIC DESIGN

Safety class systems designated Seismic Category I in Section 3.2 should consider the following seismic design guidelines.

3.7.1 Seismic Design Guidelines

3.7.1.1 Seismic Input

The seismic input to the critical facility design should be based on a detailed seismic investigation of the critical 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 Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) should be defined to be consistent with the definitions in 10 CFR 100, Appendix A.

The OBE is that earthquake which could reasonably be expected to affect the critical facility site during the operational lifetime of the facility. Considering the operational lifetime of critical facility structures, components and systems, the OBE for a long lived structure such as the reactor building may be different from the OBE for the relatively short operating life of a critical assembly. The OBE should be considered for those features of the critical facility necessary to continue operation and without undue risk to health and safety. If shutdown of the critical assembly and interruption of experiments is acceptable the OBE should be considered for those safety class systems that must function to maintain the radiological releases to within the guideline values of Chapter 15.

The SSE is considered for those structures, components and systems that must function to maintain the radiological release consequences from exceeding the guideline exposures of 10 CFR 100.

A. 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 SSE at the actual critical 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 on accompanying displacement of 36 inches, without considering soil-structure interaction.

B. Critical Damping Values

The percentage 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 plant 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. Structure 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 model 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 is used to record safety-class system response during an earthquake and permit a comparison with the design basis response. Such a comparison should be made to determine whether the facility can continue operating during and following an earthquake. Seismic instrumentation may be minimized if a critical facility is shut down immediately upon detection of a seismic event, and is not returned to operation until safety-class system integrity and operability has been verified.

3.7.4 Applicable 10 CFR General Design Criteria

It is recommended that the following 10 CFR 50, Appendix A, General Design Criterion and provisions of 10 CFR 100 be considered in the seismic design of critical 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 critical 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, "Combination of Modes and Spatial Components in Seismic Response Analysis"
6. USNRC Regulatory Guide 1.222, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components"
7. USNRC Regulatory Guide 3.14, "Seismic Design Classification for Plutonium Processing and Fuel Fabrication Plants"

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 are distributed only on 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. WASH 1361, "Safety-Related Site Parameters for Nuclear Power Plants," January 1975
5. Werner, S. D., "Study of Earthquake Input Motions for Seismic Design," TID-25438, June 1970
6. Hays, W. W., et al., "Guidelines for Developing Design Earthquake Response Spectra," CERL-TR-M-114, Army Construction Engineering Research Laboratory, U.S. Geological Survey, June 1975
7. Lomenick, T. F., "Earthquakes and Nuclear Power Plant Design," ORNL-NSIC-28, Oak Ridge National Laboratory, July 1970
8. IAEA Safety Guide SG-S1, Draft, "Earthquakes and Associated Topics for Nuclear Power Plants," International Atomic Energy Agency, Vienna
9. IAEA Safety Guide SG-S2, Draft, "Seismic Analysis and Testing of Nuclear Power Plants," *ibid.*

B. Pending Industry Standards and Codes

1. ANSI N18.4 (ANS 2.1), Proposed, "Guidelines for Determining Vibratory Ground Motion for the Design Earthquake for Nuclear Facilities"
2. ANSI N643 (ANS 2.0), Proposed, "Guidelines for Retrieval, Review, Processing, and Evaluation of Records Obtained from Seismic Instrumentation"
3. 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

Structures designated Seismic Category I should consider the seismic design guidelines of Section 3.7 and the following design guidelines for Category I structures.

3.8.1 Concrete or Steel Confinement, or Safety-Related Reactor Building

Critical facilities generally do not require reactor buildings that are designed as containments. In the event a critical facility should use a containment type reactor building, guidance, criteria and applicable standards can be found in the Design Guide for Category II Reactors, Light and Heavy Water Cooled Reactors.

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 critical 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 critical 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 and the operating basis earthquake specified for the critical facility site.
- D. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado and the safe shutdown earthquake specified for the critical facility site.
- E. Those loads to be sustained after abnormal critical 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 during normal operations and during seismic disturbances, they should help prevent the occurrence of a nuclear burst or a loss-of-fuel accident for fluid fuel systems. Loads normally applicable to critical facility building internal structures are similar to those considered in Subsection 3.8.1 for the reactor building.

Critical facility 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 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 Subsection 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 function 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 critical facility 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 critical facility 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 A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"

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

3. ANSI A138.1-1972 (ACI 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, "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 are distributed only on 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. These are normal, upset, emergency, and faulted operating conditions. Table 3.9-1 describes these operating conditions and gives typical examples of events or accidents that may correspond to each operating condition. The operating condition into which a particular event may fall will depend, to some extent, on the detailed system design and reliability; therefore, each facility should determine which postulated events and accidents correspond to each operating condition. Many factors affect the potential consequences of accidents and the frequency of their occurrence. By design, the more likely events should have lesser radiological consequences and should be accommodated within lower stress limits than the less likely accidents. The spectrum of postulated accidents to be considered in the design of critical facilities are listed in Chapter 15.

Typical critical facility mechanical systems and components that should consider the following design guidelines are moderator dump valves, control rods, control rod drive and scram mechanisms, and assembly machine separation mechanisms.

3.9.1 Design Guidelines for Safety Class Mechanical Systems and Components

The following guidelines are for the design of mechanical systems and components identified as safety class systems and components as per Section 3.2.

Critical facilities generally require few safety class mechanical components. Of those that are required, most are not covered by the ASME Boiler and Pressure Vessel Code.

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., "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.

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 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 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 Operating Basis Earthquake (OBE) should be superimposed on the loads resulting from upset events in order to determine the total upset load and the margin to the upset operating condition load limits. Likewise, the loads resulting from the Safe Shutdown Earthquake (SSE) should be superimposed

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, SHUTDOWN, ASSEMBLY OF CORE
UPSET	INCIDENTS OR OCCURRENCES OF MODERATE FREQUENCY WHICH MAY OCCUR DURING A CALENDAR YEAR AT A PARTICULAR FACILITY	LOSS OF ELECTRIC POWER, INADVERTENT REACTOR SHUTDOWN
EMERGENCY	INFREQUENT INCIDENTS OR OCCURRENCES WHICH MAY OCCUR DURING THE LIFETIME OF A FACILITY	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

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.

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 upset or faulted loads is postulated to ensure that the design of the facility is sufficiently conservative to withstand upset or faulted events that are caused by the corresponding OBE or SSE. As used herein, the term "withstand" implies that the accident consequences are within the corresponding consequence guidelines specified in Chapter 15.

3.9.2 Operability Tests for Safety Class Mechanical Components

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

3.9.3 Non-Safety Class Mechanical Components

Mechanical components designated as non-nuclear, as per Section 3.2, may be designed to non-nuclear standards if they are not required for safe shutdown of the critical assembly 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.

3.9.4 Applicable 10 CFR General Design Criteria

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

A. GDC 1, "Quality Standards and Records"

B. 10 CFR 50.55a, "Codes and Standards"

3.9.5 Standards, Codes, and Guides

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

A. Industry Standards and Codes

1. ANSI B-series standards

2. "Manual of Steel Construction," American Institute of Steel Construction."

B. RDT Standards

None

C. Guides

None

3.9.6 Supplementary Information

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

A. Background Information

1. USNRC Standard Review Plan 3.9.1, "Special Topics for Mechanical Components," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Mechanical Systems and Components," ibid.
3. USNRC Standard Review Plan 3.9.6, "Inservice Testing of Pumps and Valves," ibid.

B. Pending Industry Standards and Codes

1. ANSI N249, Proposed, "Combination of Loadings for the Design of Component Supports Under Upset, Emergency and Faulted Plan: Conditions"

3.9.7 Topics for Future Standards Development

None

B. RDT Standards

The applicability of the requirements of the following RDT standards 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 are distributed only on 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

3.10 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 Guidelines

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 the demonstration of operability by test, detail 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 principle 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 seismic qualification of Category I instrumentation and electrical equipment:

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

3.10.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the seismic qualification of 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"

- 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 1.63, "Electric Penetration Assemblies in Containment Structures for Water Cooled Nuclear Power Plants"
- 3. USNRC Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants"
- 4. USNRC Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants"

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

Typical critical facility mechanical and electrical equipment that should consider the following environmental guidelines are: moderator and/or fuel dump valves, control rods and their scram mechanisms, assembly machine separation mechanisms, portions of the protection systems, ESF systems required to mitigate postulated accidents.

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 which may result from any normal or abnormal mode of facility operation, design basis events, post-design basis events, and system tests.

The spectrum of postulated accidents at many critical facilities indicate there is no mechanism for large changes in the environmental conditions in the reactor building or fuel storage vault. As a result no special environmental design requirements are generally imposed. Those critical facilities that have normal operations or postulated accidents that result in significant changes in the reactor building environment should consider the following general requirements for environmental design and qualification.

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 based on 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 nuclear burst accident. The postulated environment should reflect a chemical environment that considers the chemical composition of all fluids and additives present in the critical assembly system or added to the reactor building environment in the course of an accident (e.g., plutonium or sodium fires, reactor building sprays, gas injection systems). The radiation environment should reflect exposures resulting from normal operation as well as postulated accidents.

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

THIS PAGE IS INTENTIONALLY LEFT BLANK

5. 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 are distributed only on 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"

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 critical facilities. Guidance pertinent to the design of these components is included in the following chapters.

3.12.2 Applicable 10 CFR General Design Criterion

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"

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 standards expand on requirements of the referenced AWS and ANS standards (listed in parentheses below). The applicability of the additional 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 subsection 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 Criteria

It is recommended that the following General Design Criterion and other provisions of 10 CFR 50 be considered in the selection and use of nuclear reactor 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 critical facility materials.

A. Industry Standards and Codes

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

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

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

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

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

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

B. RDT Standards

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-F-51079 with Modifications and Additional Requirements)

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

C. Guides

None

3.13.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or are distributed only on 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 CRITICAL 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 the public health and safety. 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 critical facilities and to the design and operational aspects that may affect decommissioning.

A. Two major alternatives for the retirement of a critical facility are:

1. 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.
2. Conversion to a New Nuclear System - Portions of a critical assembly may be disposed of by the above method. The remainder of the facility may be incorporated into a new critical assembly.

B. Critical facility design should include provisions which 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. Fluid system design to minimize the buildup of radioactive contamination.
2. Provisions for flushing contaminated systems.
3. Adequate fixed or movable shielding.
4. System design to facilitate the use of remote handling devices.
5. 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 critical facility.

3.14.2 Applicable 10 CFR General Criteria

It is recommended that the decommissioning of critical 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 critical facilities.

A. Industry Standards and Codes

None

B. RDT Standards

None

C. Guides

None

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 are distributed only on 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. AIF-NESP-009SR, "An Engineering Evaluation of Nuclear Power Reactor decommissioning Alternatives," Atomic Industrial Forum, November 1976

3. IAEA-179, "Decommissioning of Nuclear Facilities," International Atomic Energy Agency, Vienna, 1975

B. Pending Industry Standards and Codes

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

3.14.5 Topics for Future Standards Development

None

CHAPTER 4

REACTOR

4.1 CRITICAL ASSEMBLY CORE

4.1.1 Scope

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

4.1.2 Safety Function

The safety functions of the core assembly are:

A. Maintain conditions necessary to ensure that the critical assembly can be safely shut down during normal operating and accident conditions. For fluid fuel critical assemblies this function is performed jointly with the fluid system (see Chapter 5).

B. Cladding of solid fuel that forms the primary barrier against the release of fissile material, contamination and fission products from the fuel.

4.1.3 Definitions

Critical assembly core is defined as consisting of the following components:

A. Solid fuel elements or assemblies.

B. For fluid fuel systems, the portion of the critical assembly fluid system which is intended for controlled criticality.

C. In-core reactivity control systems.

D. Core support structures.

E. In-core experimental facilities.

F. Moderator and/or reflector elements.

4.1.4 Design Guidelines

4.1.4.1 Guidelines for Determining the Minimum Safety-Class of the Core Assembly Components

A. Mechanical components of the core assembly whose failure could prevent critical assembly shutdown and/or result in the uncontrolled release of

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

radioactivity to the environment should be designed to standards consistent with their importance to safety (see Sections 3.2 and 3.9).

B. If the consequences resulting from a failure of a component in the core assembly does not exceed the radiological consequence guidelines of Chapter 15, and the resulting damage is within the guideline of this Chapter, the components may be non-nuclear.

4.1.4.2 Guidelines for the Design of the Core Assembly

A. For normal and upset facility conditions (defined in Table 3.9-1, the critical facility should be designed such that fuel cladding damage and/or fuel melting (for solid plutonium or uranium fuel) or fuel boiling and/or fuel burning (fluid systems) is not expected 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 critical facility should be designed such that any fuel or cladding damage that occurs should not prevent critical assembly shutdown.

C. Potential effects of experimental facilities and handling equipment associated with a critical assembly core should be considered in the design of the core assembly. The experimental facilities should be designed such that any 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. To permit contact maintenance solid plutonium fuel elements or assemblies should normally be clad to minimize health hazards from plutonium contamination. A suitable paint coating may be used for uranium fuel elements. This does not preclude critical experiments using bare plutonium and uranium fuel when appropriate handling precautions are instituted.

F. A source of neutrons should be available to produce a meaningful indication of neutron multiplication during any approach to criticality. This source of neutrons may be inherent to the core material or provided by an appropriately positioned external neutron source.

G. Instrumentation should be provided to monitor critical assembly 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.

H. The core assembly should be designed with inherently large (and if practical, prompt) negative temperature coefficients and with limited excess reactivity in order to suitably limit the consequences of potential reactivity accidents.

I. The core assembly should be protected from the effects of severe natural and man-made phenomena (see Chapter 3). If bare plutonium is employed in a core assembly, the facility design may require additional structures such as cells, hoods, gloveboxes or barriers to assure adequate protection of the core assembly from the effects of the above phenomena.

J. Mechanical components of the core assembly should consider the guidance of Section 3.9.

4.1.4.3 Maintenance, Testing, and Inspection

A. Preoperational testing is described in Chapter 14.

4.1.4.4 Interfaces with Other Systems

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

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

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

4.1.5 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 be considered in the design of a core assembly:

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

4.1.6 Standards, Codes, and Guides:

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

- A. Industry Standards and Codes

1. N405-1975 (ANS-1), "Safety Guide for the Performance of Critical Experiments"

2. N18.3-1972, "Fuel Element Identification"

B. RDT Standards

None

C. Guides

None

4.1.7 Supplementary Information

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

A. Background Information

None

B. Pending Industry Standards and Codes

1. 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 critical assembly core reactivity to ensure that the critical assembly may be safely shutdown and that the fuel damage limits of Section 4.1 are not exceeded for any facility condition.

B. Limit the damage to the critical facility systems that may result from a nuclear burst 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 or reflecting systems or variable moderating systems which must function in order to assure safe operation and shutdown of the critical assembly, including control rods, movable reflector elements, moderator and fuel fill and dump systems, and poison injection systems.

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 of Safety Class 3 if their failure prevents safe shutdown of the critical assembly, or results in consequences in excess of the core damage guidelines of Section 4.1 or 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 shutdown of the critical assembly and resulting consequences are within the core damage guidelines of 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.

Figure 4.2-1. Guide to the use of section 4.1 (continued).

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, diverse, and fail safe means of reactivity control should be provided. These may include control rods, movable reflector elements, moderator or fuel dumps, poison injection, or other suitable means of reactivity control.

C. Both reactivity control systems should be capable of shutting down an operating critical assembly 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 reactivity safety features such as a large (and in some facilities, prompt) negative 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 positive and negative reactivity addition rates required for the reactivity control system.

F. The critical assembly and reactivity control system should accommodate upset conditions (defined in Table 3.9-1) and facilitate a rapid return to normal conditions.

G. 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.

H. 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.

I. Mechanical portions of the reactivity control system should consider the design guidance of Section 3.9.

4.2.4.3 Maintenance, Testing, and Inspection

A. The reactivity control system should be designed to permit the performance of periodic functional tests of the entire system, including tests necessary for determining 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 Section 7.1 apply to the 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. Critical Assembly Core
- B. Fluid Fuel System
- C. Protection System
- D. Non-safety I&C System
- E. Class IE Electric Power System
- D. Non-Class IE Electric Power System
- G. 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. GDC 10, "Reactor Design"
- B. GDC 13, "Instrumentation and Control"
- C. GDC 20, "Protection System Functions"
- D. GDC 24, "Separation of Protection and Control Functions"
- E. GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"

- F. GDC 26, "Reactivity Control System Redundancy and Capability"
- G. GDC 27, "Combined Reactivity Control Capability"
- H. GDC 28, "Reactivity Limits"
- I. GDC 29, "Protection Against Anticipated Operational Occurrences"

4.2.6 Standards, Codes, and Guides

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 Control

1. ANSI N16.1-1975 (ANS 8.1), "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"
2. ANSI N16.4-1971 (ANS 8.3), "Use of Borosilicate Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material"
3. ANSI N16.8-1975 (ANS 8.10), "Criteria for Nuclear Criticality Safety Controls in Operations Where Shielding Protects Personnel"
4. ANSI N16.9-1975 (ANS 8.11), "Validation of Calculational Methods of Nuclear Criticality Safety"
5. ANSI N405-1975 (ANS-1), "Safety Guide for the Performance of Critical Experiments"

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

A. Background Information

None

B. Pending Industry Standards and Codes

1. ANSI N16.6 (ANS 8.8), Draft, "Criticality Safety Limits for Special Applications"

2. ANSI N16.11 (ANS 8.13.1), Draft, "Evaluation of Neutron Interactions Between Units of Low U-235 Enrichment"

3. ANSI N16.12 (ANS 8.14), Draft, "Neutron Absorbers in Solutions of Fissile Material"

4. ANSI N16.13 (ANS 8.13.2), Draft, "Guide for Evaluating Interaction Between Units of Low Enriched Uranium Using the Surface Density Method"

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

6. ANSI/ANS 8.13.1, Draft, "Guide for Evaluating Interaction Among Units of Low Enriched Uranium by the Solid Angle Method"

4.2.8 Topics for Future Standards Development"

None

CHAPTER 5

FLUID FUEL SYSTEM

5.1 Scope

This section provides guidelines for the design of a fluid fuel critical facility and for determining the minimum safety class of components of the system. Figure 5.1-1 is a general guide to the use of this chapter.

5.2 Safety Function

The safety functions of the fluid fuel system are:

A. To provide a barrier against the release of toxic materials and radioactivity to the environment, if such a release may result in exposure in excess of applicable guidelines.

B. To assure that fuel design limits of Section 4.1 are not exceeded during any condition of routine operation (which includes normal and upset facility conditions).

C. Reactivity control of the critical assembly.

5.3 Definition

The critical facility fluid system is defined as those components which form the primary boundary, which includes the following:

A. Vessel or device used to contain the fluid fuel in the critical assembly core. This includes any penetrations of the container for reactivity control devices or other experimental systems or apparatus.

B. Pumps, valves, and piping completing the fluid flow paths for adding or removing fuel from the core and transferring it to the fuel handling and storage facilities.

C. Cover gas volume.

D. Other attached piping out to a suitable boundary.

5.4 Design Guidelines

5.4.1 Guidelines for Determining the Minimum Safety Class of the Fluid Fuel System

If a failure of the primary boundary can result in unacceptable nuclear, mechanical, or thermal consequences, or, if the off-site doses exceed 0.5 rem, the

Figure 5.1-1. Guide to the use of chapter 5.

Figure 5.1-1. Guide to the use of chapter 5 (continued).

portions of the primary boundary to prevent such failure should be a minimum of SC-3.

If the nuclear, mechanical, or thermal consequences are acceptable and off-site doses are less than 0.5 rem, the primary boundary may be designated non-nuclear.

5.4.2 Guidelines for the Design of a Fluid Fuel System

A. The fluid system primary boundary should be designed, fabricated, erected and tested so as to have a low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

B. Fluid fuel systems should contain appropriate design features (e.g., raschig rings, geometry) to prevent inadvertent criticality.

C. The fluid system should be designed with appropriate margin to assure that specified design conditions of the primary boundary are not exceeded during any condition of normal operation or upset occurrences.

D. Means should be provided to detect and collect leakage from the primary boundary.

E. The primary boundary should be protected against the effects of severe natural and man-made phenomena (see Chapter 3).

F. Mechanical components of the fluid system should consider the design guidance of Section 3.9.

G. The environmental conditions of Section 3.11 should be considered in the design of fluid system components.

5.4.3 Maintenance, Testing, and Inspection

A. The fluid fuel system should be designed to facilitate periodic inspections and necessary maintenance within the radiological guidelines for normal operation.

5.4.4 Interfaces With Other Systems

The fluid fuel system may interface with other facility systems, including the following:

- A. Critical Assembly Core
- B. Reactivity Control System
- C. Fuel Handling and Storage System
- D. Reactor Building

- E. Protection System
- F. Non-Safety I&C Systems
- G. Non-Class IE Electric Power System
- H. Experimental Facilities
- I. Radioactive Waste System

5.5 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, "Licensing of Production and Utilization Facilities," be considered in the design of a critical facility fluid fuel system.

- A. GDC 10, "Reactor Design"
- B. GDC 14, "Reactor Coolant Pressure Boundary"
- C. GDC 15, "Reactor Coolant System Design"
- D. GDC 30, "Quality of Reactor Coolant Pressure Boundary"
- E. GDC 32, "Inspection of Reactor Coolant Pressure"

5.6 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 a reactor fluid system.

- A. Industry Standards and Codes
None
- B. RDT Standards
None
- C. Guides
None

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

A. Background Information

None

B. Pending Industry Standards and Codes

1. ANSI N16.7, "Nuclearly Safe Pipe Intersections for Solutions of Fissionable Materials"

2. ANSI N16.10, "Nuclear Criticality Control and Safety of Pu-U Fuel Mixtures Outside Reactors"

5.8 Topics for Future Standards Development

None

CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 REACTOR BUILDING

As defined in Section 1.5 a reactor building is that structure whose primary function is to house the critical assembly and associated experimental systems during the performance of critical experiments. It may also house a fissile material storage area and auxiliary equipment to support operations.

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 containment, confinement, safety-related reactor building or non-safety-related reactor building. Basic guidance relevant to the design of confinements, safety-related reactor buildings, and non-safety-related reactor buildings is provided in this section. Generally critical facilities do not require a containment. In the event a critical facility should use a containment, design guidelines can be found in the Design Guide for Category II Reactors Light and Heavy Water Cooled Reactors. 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 radioactive and toxic fissile material to the environment such that the potential radiological consequences are maintained within the guidelines for the appropriate operating condition (see Chapter 15).

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

6.1.3 Definitions

The various types of reactor buildings are defined below:

A. Containment

A containment is defined as those structures and components (including the pressure retaining containment shell, isolation valves, heat removal systems, and cleanup systems) which enclose the critical assembly and associated systems and must function in order to provide a low leakage, pressure retaining barrier against the uncontrolled release of radioactive and toxic fissile materials to

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 AND TOXIC FISSILE MATERIAL TO THE ENVIRONMENT	PROTECT ENCLOSED SYSTEMS FROM NATURAL PHENOMENA AND MISSILES	
CONTAINMENT	YES	YES	LOW-LEAKAGE, PRESSURE RETAINING SEISMIC CATEGORY I STRUCTURE
CONFINEMENT	YES	YES	DIRECTED-LEAKAGE SEISMIC CATEGORY I STRUCTURE (MAINTAINED AT A NEGATIVE PRESSURE) WITH A FILTERED EX- HAUST STREAM
SAFETY RELATED REACTOR BUILDING	NO	YES	SEISMIC CATEGORY I STRUCTURE
NON-SAFETY RELATED REACTOR BUILDING	NO	NO	INDUSTRIAL TYPE BUILDING

the environment and which additionally protect enclosed systems against the effects of severe natural and man-made phenomena. A containment is generally not required for critical facilities.

B. Confinement

A confinement is defined as those structures and components (including the building structure, ventilation paths, filtration and absorption units and fans) which enclose the critical assembly and associated systems and must function in order to control the release of radioactivity or toxic material to the environment. The building is usually maintained at a slight negative pressure. Its exhaust path is directed through the appropriate equipment necessary to reduce 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.

C. A safety-related reactor building is defined as those structures which enclose and must protect the critical assembly and associated systems from the effects of severe natural and man-made phenomena, but which are not intended for controlling the release of radioactive or toxic materials to the environment.

D. Non-Safety Related Reactor Building

A non-Safety related reactor building is defined as those structures which enclose the critical assembly and associated systems, but neither control the release of radioactive or toxic materials 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 containment, confinement, safety-related reactor building or non-safety reactor building to house critical assemblies. The location of the reactor building, in particular its distance from the site boundary, and other critical facility and on-site facilities is an important consideration in selecting the type of reactor building. The need for a containment or confinement will be based on the potential radiological impact of the facility. The need for providing protection to enclosed systems or components against the effects of severe natural 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 6.1.4.7.

A. Containment and Confinement

The reactor building should be designed as a containment or confinement if the radiological consequence guidelines for the appropriate facility operating

TABLE 6.1-2. MINIMUM SAFETY CLASS RECOMMENDATIONS FOR REACTOR BUILDINGS

REACTOR BUILDING TYPE	POTENTIAL RADIOLOGICAL CONSEQUENCES OF ACCIDENTS ⁽¹⁾	
	THE RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS WOULD EXCEED THE GUIDELINES FOR THE APPROPRIATE FACILITY OPERATING CONDITION IF CREDIT WERE NOT TAKEN FOR THE CONTAINMENT OR CONFINEMENT CAPABILITY OF THE REACTOR BUILDING	THE RADIOLOGICAL CONSEQUENCES OF POSTULATED ACCIDENTS WOULD NOT EXCEED THE GUIDELINES FOR THE APPROPRIATE FACILITY OPERATING CONDITION IF THE BUILDING WERE NOT PROVIDED
CONTAINMENT	SC-2 OR -3 ⁽²⁾	--
CONFINEMENT	SC-2 OR -3 ⁽²⁾	--
SAFETY-RELATED REACTOR BUILDING	--	SC-3 ⁽³⁾
NON-SAFETY RELATED REACTOR BUILDING	--	NN

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

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

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

condition (see Chapter 15) cannot be met without taking credit for some design feature of the reactor building in order to control the release of radioactivity to the environment. The selection of containment or confinement should be based on the degree of effluent concentration reduction required. Both containment and confinement provide 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 demonstrates an inherent degree of safety due to reduced source terms or other features that permit operation without a containment or confinement capability while maintaining the radiological consequences of postulated accidents within the guidelines of the appropriate facility operating condition (see Chapter 15).

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.

6.1.4.2 Guidelines for the Design of a Containment and Associated Systems

A. The Containment Structure

See the Design Guide for Category II Reactors.

6.1.4.3 Guidelines for the Design of a Confinement and Associated Systems

A. Confinement Structure and Exhaust System

1. The potential sources of energy release to the confinement should be considered. Such items as core nuclear burst, static heat sources and sinks, potential fire sources (e.g., plutonium, liquid metal), combustible gas reactions and other possible energy sources and uncertainties that may necessitate additional design margin.

2. The potential releases of radioactivity and toxic fissile materials 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 and on-site doses resulting from postulated accidents.

3. 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.

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

5. When active components are provided to effect confinement system configuration and/or exhaust flow a single active failure should not result in unacceptable radiological consequences.

6. Adequate instrumentation should be provided to actuate active components of the confinement system when necessary or to notify the operator of conditions requiring manual isolation. Such instrumentation should consider the design guidance of Section 7.1 for protection system.

7. 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.

8. The confinement should provide protection to enclosed systems from the effects of severe natural and man-made phenomena.

9. The confinement should be designated Seismic Category I as per Section 3.2 and consider the design guidance of Section 3.8 for Seismic Category I Structures.

10. Mechanical portions of the confinement system should consider the design guidance of Section 3.9.

11. The confinement system should be capable of performing its safety functions under the environmental conditions specified in Section 3.11.

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

A. The safety-related reactor building should be designed Seismic Category I as per Section 3.2 and consider the design guidance of Section 3.8 for Seismic Category I Structures.

B. The safety-related reactor building should provide protection to enclosed systems from the effects of severe natural and man-made phenomena.

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

A. The general design criteria in ERDA Manual Chapter 6301 should be considered in the design of a non-safety-related reactor building.

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

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

A. The performance of periodic functional tests of the complete containment or 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 Guides 1.52 and 1.140.

6.1.4.7 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. Class IE Electric Power System
- D. Non-Class IE Electric Power System
- E. Fuel Handling and Storage System
- F. Fire Protection System
- G. Experimental Facilities
- H. Radioactive Waste System
- I. 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. Confinement

- 1. GDC 60, "Control of Releases of Radioactive Materials to the Environment"
- 2. GDC 64, "Monitoring Radioactivity Releases"

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

None

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 building.

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 N509-1976, "Nuclear Power Plant Air Cleaning Units and Components"
- c. ANSI N510-1975, "Testing of Nuclear Air Cleaning Systems"
- d. ANSI/UL 586-1977, "Safety Standard for Test Performance of High Efficiency Particulate Air Filter Units"

2. Safety-Related Reactor Buildings

None

3. Non-Safety-Related Reactor Buildings

- 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. "Specification for Design, Fabrication and Erection of Structural Steel Buildings," American Institute of Steel Construction, 1969
- f. "Code of Standard Practice for Steel Buildings and Bridges," American Institute of Steel Construction, 1976
- g. ERDAM 6301, "General Design Criteria"

B. RDT Standards

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

1. Confinements

a. E9-1T, "HEPA Filters"

2. Safety-Related Reactor Buildings

None

3. Non-Safety-Related Reactor Buildings

None

C. Guides

1. Confinements

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

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

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

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

None

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

A. Background Information

1. USNRC Standard Review Plan 6.5.1, "ESF Filter systems," *ibid.*

2. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.

3. Burchsted, C. A., et al., "Nuclear Air Cleaning Handbook" ERDA 7621, Oak Ridge National Laboratory, March 31, 1976

4. MIL-F-51068D, "Filter Particulate High Efficiency Fire Resistant"

5. "Leakage Characteristics of Openings for Reactor Housing Components" NAA-SR-MEMO-5137, Atomics International, June 20, 1960

6. Koontz, R. L., et al., "Conventional Buildings for Reactor Containment," NAA-SR-10100, Atomics International, July 25, 1965

7. Koontz, R. L., et al., "Leakage Characteristics of Conventional Building Components for Reactor Housing Construction," Trans. Am. Nucl. Soc., November 1961

8. "A Guide to Good Practices at Plutonium Facilities," BNWL-2086, September 1977

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 SYSTEMS

6.2.1 Scope

This section provides guidelines for determining the need for an Engineered Safety Feature (ESF) ventilation system, including a control room habitability system. In addition, basic guidance relevant to the design of an ESF ventilation system is provided. Figure 6.2-1 is a guide to the use of this section.

6.2.2 Safety Function

The safety functions of the ESF ventilation system are to:

A. Protect on-site personnel in the control room and in other normally occupied enclosed areas of the critical facility (e.g., experiment areas) from excessive exposure to toxic agents or exposure to airborne radiation that would result in doses in excess of the guidelines for the appropriate facility operating condition (see Chapter 15) and/or

B. Maintain the environmental conditions of essential areas of the critical facility within acceptable limits for human habitability and/or for the operation of safety class equipment.

6.2.3 Definition

The ESF ventilation system is defined as the heating, ventilation, and air conditioning systems (including associated duct work, blowers, fans, filter and adsorber systems, heat exchangers, and valves) which must function in order to protect the control room operators and other personnel in normally occupied enclosed areas of the reactor facility from toxic gas and airborne radiation hazards. In addition, this system must function in various areas of the facility in order to maintain the environment within acceptable limits.

6.2.4 Design Guidelines

6.2.4.1 Guidelines for Determining the Need for and Minimum Safety Class of an ESF Ventilation System

A. An ESF ventilation system should be provided to control potential airborne radiation and toxic gas hazards that would result in exposure to on-site personnel in enclosed spaces in excess of the following guidelines.

1. Airborne Radiation Hazard

Maximum potential dose to personnel should not exceed the guidelines for the appropriate facility operating condition (see Chapter 15).

In determining the on-site dose consequences that may result from airborne radiation hazards, the site emergency plan and the ability to evacuate and transport on-site personnel should be considered.

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

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

2. Toxic Gas

Potential toxic gas exposures should not result in chronic effects from exposure, and acute effects, if any, should be reversible within a short period of time without benefit of medication other than the use of self-contained breathing apparatus.

B. An ESF ventilation system may be required to maintain the temperature, pressure, and humidity environment within limits for human habitability and/or within the qualification limits of safety class equipment if those limits would be exceeded during normal operation or accident conditions.

C. If required for safety, the ESF ventilation system is a minimum of Safety Class 3.

6.2.4.2 Guidelines for the Design of an ESF Ventilation System

A. This system should be capable of performing the required safety functions on off-site or on-site electric power (assuming one is unavailable).

B. The ESF ventilation system should be protected against the effects of severe natural and man-made phenomena (see Chapter 3).

C. Mechanical components of the ESF ventilation system should consider the design guidance in Section 3.9.

D. The selection of the type of protective ventilation system required for protection of personnel should be based on the magnitude and nature of the potential hazard. Several alternative types of protective ventilation systems are listed below.

1. Zone isolation, with the incoming air filtered with positive pressure maintained by fans.

2. Zone isolation, with filtered, recirculated air.

3. Zone isolation, with filtered, recirculated air and positive pressure maintained in the zone.

4. Dual, widely separated air intakes, with the contamination-free intake providing positive pressure in the zone.

5. Bottled air supply for occupants of the zone.

E. The air filtration units should be designed to remain functional throughout postulated accidents and should retain collected radioactive material after the accident.

F. The air filtration units should be provided with appropriate radiation shielding to maintain occupational doses within the guidelines of Chapter 12.

G. The system should incorporate sufficient redundancy such that the system safety function can be accomplished with a single failure of an active component.

H. Each filter train should be able to prefilter the air, to remove moisture ahead of any charcoal adsorbers, and remove particulate matter before and after a charcoal adsorber.

I. The ESF ventilation system should be environmentally qualified to withstand the environmental conditions during normal operations and during the spectrum of accidents postulated in Chapter 15 to ensure that the system safety functions will be completed when required (see Section 3.11).

6.2.4.3 Maintenance Testing and Inspection

A. The ESF ventilation system should be designed to permit the performance of periodic functional tests of the complete system.

B. Provisions should be included for the visual inspection of the system and associated components.

C. Qualification testing, maintenance, and inplace testing of filtration and adsorption units should meet the guidelines of USNRC Regulatory Guide 1.52.

6.2.4.4 Interfaces With Other Systems

The ESF ventilation system may interface with the following other facility systems.

- A. Reactor Building
- B. Protection System
- C. Class IE Electric Power System
- D. Fuel Handling and Storage System
- E. Fire Protection System
- F. Radioactive Waste System
- G. Radiation Protection System

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

6.2.5 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 engineered safety feature ventilation systems.

A. GDC 19, "Control Room"

6.2.6 Standards, Codes, and Guides

In addition to the general standards, codes, and guides listed in Chapter 3, it is recommended that the following be considered in the design of an engineered safety ventilation system.

A. Industry Standards and Codes

1. ANSI N101.1-1972, "Efficiency Testing of Air-Cleaning Systems Containing Devices for Removal of Particles"
2. ANSI N509-1976, "Nuclear Power Plant Air-Cleaning Units and Components"
3. ANSI N510-1975, "Testing of Nuclear Air-Cleaning Systems"
4. ANSI/UL 586-1977, "Safety Standard for Test Performance of High Efficiency Air Particulate Units"

B. RDT Standards

The applicability of the requirements of the following RDT standards 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 Atmosphere Cleanup System Air Filtration and Adsorption Units for Light Water Cooled Nuclear Power Plants"
2. USNRC Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"
3. USNRC Regulatory Guide 3.2, "Efficiency Testing of Air-Cleaning Systems Containing Devices for Removal of Particles"
4. USNRC Regulatory Guide 3.7, "Monitoring of Combustible Gases and Vapors in Plutonium Processing and Fuel Fabrication Plants"
5. USNRC Regulatory Guide 3.12, "General Design Guide for Ventilation Systems of Plutonium Processing and Fuel Fabrication Plants"

6.2.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and

standards that are proposed for future development or are distributed only on a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 6.4, "Habitability System," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 6.5.1, "ESF Filter Systems," *ibid.*
3. USNRC Standard Review Plan 9.4.5, "Engineered Safety Feature Ventilation System" *ibid.*
4. "Heating, Ventilation and Air-Conditioning Guide," American Society of Heating and Ventilating Engineers, 1953
5. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.
6. Burchsted, C. A., et al., "Nuclear Air Cleaning Handbook," Oak Ridge National Laboratory, March 31, 1976
7. Griwatz, G. H., Friel, J. V., and Bicehouse, J. L., "Entrained Moisture Separators for Fine (1-10u) Water-Air-Steam Service: Their Performance, Development, and Status,": MSAR 71-45, Mine Safety Appliances Research Corporation.
8. WASH-1287, "Manual of Respiratory Protection Against Radioactive Material," USAEC, 1974
9. Murphy, K. G. and Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974
10. MIL-F-51068D, "Filter, Particulate, High Efficiency, Fire Resistant, 1974
11. ERDAM Appendix 6301, "General Design Criteria," Part II, Section 1, Plutonium Facilities
12. "A Guide to Good Practices at Plutonium Facilities," BNWL-2086, September 1977

B. Pending Industry Standards and Codes

1. ANSI N189 (ANS 59.2), Proposed, "Safety Related Ventilation Systems"
2. ANSI N290, "Design, Construction, and Operation of Ventilation Systems for Mixed Oxide (U, Pu) Fuel Fabrication Plants"

6.2.8 Topics for Future Standards Development

The following topical listing identifies areas in which the development of additional nuclear standards may result in a consistent approach to critical facility design and modification.

A. Guidelines for providing protection of personnel at DOE critical facilities against the hazards that may be created by experimental reactor systems and/or experimental operational anomalies or maintenance activities (e.g., toxic gases and airborne radiation hazards to operators and/or experimenters inside the reactor building).

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 class 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, operate the critical facility 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 (from measured process variables to the input terminals of the devices that perform the protective action) which generate those signals involved with the protective functions. Protective function signals include those that initiate reactor trip or the operation of engineered safety feature systems (see Chapter 6).

B. Safety class instrumentation includes displays of critical 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 critical facility 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 critical

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

assembly 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 critical assembly fluid system boundary (Section 5.1).
- D. Damage guidelines for a safety class reactor building (Section 6.1).
- E. Radiation and toxic gas exposure guidelines for facility operators (Section 6.2 and Chapters 12 and 15).
- F. Off-site radiological consequence guidelines (Chapters 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 SC-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 critical assembly 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 critical 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 functions 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. Return to operation requires subsequent deliberate operator action.

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

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

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

J. Protection system instrumentation and controls sufficient for placing the critical facility in a safe shutdown condition should be provided in a location remote from the control room for use during any condition which may make the control room uninhabitable. 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 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 critical assembly 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 and calibration of protection system instrument channels and interlocks.

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

7.1.4.4 Interfaces With Other Systems

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

- A. Critical Assembly Core
- B. Reactivity Control System
- C. Fluid Fuel System
- D. Reactor Building
- E. ESF Ventilation System
- F. Non-Safety I&C System
- G. Class IE Electric Power System
- H. Fuel Handling and Storage System
- I. Experimental Facilities

J. 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 29, "Protection Against Anticipated Operational Occurrences"
- J. GDC 64, "Monitoring Radioactivity Releases"
- K. 10 CFR 50.55a, "Codes and Standards"

7.1.6 Standards, Codes, and Guides

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

- A. Industry Standards and Codes
 - 1. IEEE 279-1971 (ANSI N42.7-1972), "Criteria for Protection Systems for Nuclear Power Generating Stations"
 - 2. IEEE 336-1977 (ANSI N45.2.4), "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 for Reliability Analyses of Nuclear Power Generating Stations"

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

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

7. IEEE 420-1972 (ANSI N41.17), "Class IE Control Switchboards for Nuclear Power Generating Stations"

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

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

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

11. ANSI N405-1975 (ANS 1), "Safety Guide for the Performance of Critical Experiments"

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

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

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

5. 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 1.62, "Manual Initiation of Protective Action"

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

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

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

7.1.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or are distributed only on 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 7.3, "Engineered Safety Feature Systems," *ibid.*

4. USNRC Standard Review Plan 7.4, "Systems Required for Safe Shutdown," *ibid.*

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

6. USNRC Standard Review Plan 7.6, "All Other Instrumentation Required for Safety," *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-111, Oak Ridge National Laboratory, July 1973

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

13. IAEA Safety Guide SG-D3, Draft, "Protection Systems in Nuclear Power Plants"

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

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

B. Pending Industry Standards and Codes

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

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

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

4. ANSI N297, "Design Basis and Periodic Testing of Protection System"

7.1.8 Topics for Future Standards Development

None

7.2 NON-SAFETY CLASS INSTRUMENTATION AND CONTROL SYSTEM

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 must not prevent the interfacing protection system from completing its safety function when required.

7.2.3 Definition

Non-safety class instrumentation and control system 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 System

A. Non-safety class I&C systems may receive outputs 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.

7.2.4.2 Maintenance, Testing, and Inspection

A. 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.

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

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. Fluid Fuel System
- C. Reactor Building
- D. Protection System
- E. Non-Class IE Electric Power System
- F. Fuel Handling and Storage System
- G. Fire Protection System
- H. Experimental Facilities
- I. 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"

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 system.

- 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 384-1977 (ANSI N41.14), "Criteria for Independence 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 are distributed only on a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 7.7, "Control Systems Not Required for Safety," NUREG 75/087, September 1975
2. Harrer, J. M., Beckerly, J. G., "Nuclear Power Reactor Instrumentation Systems Handbook," Volumes 1 and 2, TID-25952-P1 and -P2, USAEC 1973 and 1974
3. "Safe Operation of Critical Assemblies and Research Reactors," IAEA Safety Series No. 4, International Atomic Energy Agency, Vienna, 1961

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 I Reactors, Critical Facilities, 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 critical assembly in a safe shutdown condition.

B. No active, electric powered systems are required for shutdown cooling of the critical assembly.

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 critical facility, that facility should refer to the Design Guide for Category II Reactors Light and Heavy Water Cooled 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.

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 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. Fluid Fuel System

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

- C. Non-Safety Instrumentation and Control Systems
- D. Fuel Handling and Storage System
- E. Fire Protection System
- F. Experimental Facilities
- G. 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"

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 C2-1977, "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. New or spent fuel to be transported from a critical facility should be prepared for shipment in accordance with the provisions of ERDA Manual Chapter 0529, "Safety Standards for the Packaging of Fissile Material and Other Radioactive Material." 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, process, mix and store new and irradiated fuel in a manner which assures:

- A. No inadvertent criticality during handling, processing, mixing or storage.
- B. Minimize leakage from fluid systems.
- C. 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 includes those components, systems and structures including containers, racks, vaults, hoods, gloveboxes, tanks, piping, valves, pumps and sumps required for performing the following functions:

- A. Receiving fuel from a carrier and making necessary accountability inspections.
- B. New fuel storage prior to use in the critical assembly.
- C. Handling, processing, or mixing fuel during movement between storage areas and the critical assembly 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.

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

9.1.4 Design Guidelines

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

A. Fluid system 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 excess of 0.5 rem or (2) they are required for maintaining new or spent fuel in a subcritical configuration.

B. Portions of the fuel handling and storage system that pass through a safety class structure should be designed to a safety class that is compatible with the structure.

C. 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. Fluid fuel additions and withdrawals from a critical assembly should be of such design that all essential operations are capable of remote operation. Operations such as fluid fuel preparation, mixing, chemical separation, transfer, sampling, and storage may be performed manually to improve their efficiency and reliability.

C. 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).

D. Mechanical portions of the handling and storage equipment should consider the design guidance in Section 3.9.

E. Buildings containing liquid fuel systems should be provided with floor drain, sump, and collection systems designed to maintain any leakage accumulations during normal operations and following any postulated accidents in subcritical configurations (for example, by a catch basin of Raschig Rings or geometry).

F. 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).

G. Fluid fuel systems containing plutonium or bare solid plutonium that is vented and/or exposed directly to the atmosphere should be enclosed by a hood, glovebox, or other such device so that plutonium contamination from normal use and vapors resulting from fires are contained such that personnel exposures are maintained within guidelines (see Chapter 12). This is not intended to preclude

the use of bare solid plutonium in air for research and experimental purposes when appropriate health physics and safety precautions are taken.

H. Hoods, gloveboxes or other enclosures for containing plutonium contamination should be kept at slightly negative pressures.

I. Fuel handling and storage systems which contain bare plutonium that is directly exposed to the atmosphere or enclosed by hoods or gloveboxes should have inert gas injection systems to smother potential plutonium fires.

J. Radioactive gas concentrations, including airborne radioactive materials 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.

K. Lifting and transport equipment of the fuel handling system should be designed to prevent dropping a fuel assembly, or prevent spillage or leakage from fluid fuel containers.

L. 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.

M. Provisions should be made to ensure that the critical assembly 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.

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. Critical Assembly Core Components
- B. Reactivity Control System
- C. Fluid Fuel System
- D. Reactor Building
- E. Engineered Safety Feature Ventilation System
- F. Protection System

- G. Non-Safety I&C System
- H. Non-Class IE Electric Power System
- I. Fire Protection System
- J. Radioactive Waste System
- K. 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 N16.1-1975, "Nuclear Criticality Safety in Operations With Fissile Materials Outside Reactors"
2. ANSI N16.4-1971, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material"
3. ANSI N16.5-1975, "Nuclear Criticality Safety in the Storage of Fissile Materials"
4. ANSI N16.8-1975 "Criteria for Nuclear Criticality Safety Controls in Operations Where Shielding Protects Personnel"

5. ANSI N16.9, "Validation of Computational Methods for Nuclear Criticality Safety"

6. ANSI N405-1975 (ANS 1), "Safety Guide for the Performance of Critical Experiments"

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

B. RDT Standards

None

C. Guides

1. Safety Class Components

a. USNRC Regulatory Guide 5.8, "Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Drying and Fluidized Bed Operations"

b. USNRC Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants"

2. Non-Safety Class Components

None

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

A. Background Information

1. USNRC Standard Review Plan 9.1.4, "Fuel Handling System," *ibid.*

2. Branch Technical Position APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants," (attached to SRP 9.1.4), *ibid.*

3. ERDAM Appendix 6301, General Design Criteria, Part II, Section 1, "Plutonium Facilities"

4. "A Guide to Good Practices at Plutonium Facilities," BNWL-2086, September 1977

B. Pending Industry Standards and Codes

1. ANSI N369 (ANS 11.16), Draft, "Gloveboxes"

2. ANSI N16.6 (ANS 8.8), Draft, "Criticality Safety Limits for Special Applications"

3. ANSI N16.7 (ANS 8.9), Draft, "Nuclear Safe Pipe Intersections for Solutions of Fissile Materials"

4. ANSI N16.10 (ANS 8.12), Draft, "Nuclear Criticality Control and Safety of Pu-U Fuel Mixtures Outside Reactors"

5. ANSI N16.11 (ANS 8.13), Draft, "Evaluation of Neutron Interactions Between Units of Low ²³⁵U Enrichment"

6. ANSI N16.12 (ANS 8.14), Draft, "Neutron Absorbers in Solutions of Fissile Material"

7. ANSI N16.13 (ANS 8.13.2), Draft, "Guide for Evaluating Interaction Between Units of Low Enriched Uranium Using the Surface Density Method"

8. ANSI N347, Proposed, "Criteria for the Location, Design, Construction, Operations, and Maintenance of Facilities"

9.1.8 Topics for Future Standards Development

None

THIS PAGE IS INTENTIONALLY LEFT BLANK

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 combination 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 plutonium fire.

9.2.3 Definitions

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

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

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. Extinguishing systems used within the reactor building or fissile material storage areas should not result in flooding the critical assembly, fuel in storage or engineered safety features. Such flooding could result in an inadvertent criticality or reduced ESF effectiveness (e.g., building ventilation filters).

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. Mechanical portions of the fire protection system should consider the design guidance in Section 3.9.

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

1. Non-combustible 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.

9.2.4.3 Maintenance, Testing, and Inspection

A. 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. Engineered Safety Feature Ventilation System
- C. Protection System
- D. Non-Safety Instrumentation and Control System
- E. Class IE Electric Power System
- F. Non-Class IE Electric Power System
- G. Fuel Handling and Storage System
- H. 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 N18.10 (ANS 59.4), Draft, "Fire Protection Criteria for Safety Related Structures and Equipment for Water Cooled and Moderated Nuclear Power Generating Plants"

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

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

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

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

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

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

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

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

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

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

r. NFPA 22-1974, "Water Tanks"

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

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

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

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

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

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

y. NFPA 214-1971, "Water Cooling Towers"

2. Fire Protection System Administration

a. ANSI N45.2.3-1975, "Housekeeping During the Construction Phase of Nuclear Power Plants"

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

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

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

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

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

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

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

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

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

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

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

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

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

o. 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"

3. USNRC Regulatory Guide 3.7, "Monitoring of Combustible Gases and Vapors in Plutonium Processing and Fuel Fabrication Plants"

4. USNRC Regulatory Guide 3.16, "General Fire Protection Guide for Plutonium Processing and Fuel Fabrication 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 are distributed only on a limited basis as preliminary drafts are listed for information only.

A. Background Information

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

2. NUREG-0050, "Recommendations Related to Brown's Ferry Fire," USNRC, February 1976

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

4. ERDAM Appendix 6301, General Design Criteria, Part II, Section 1, "Plutonium Facilities"

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

6. IAEA Safety Guide SG-D2, Draft, "Safety Guide for Fire Protection in Nuclear Power Plants"

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

B. Pending Industry Standards and Codes

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

2. ANSI N280, "Fire Protection for Pu Fuel Fabrication Facilities"

9.2.8 Topics for Future Standards Development

None

THIS PAGE IS INTENTIONALLY LEFT BLANK

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 critical assembly core and those that penetrate and are a part of the primary boundary of fluid fueled critical facilities. The ERDA Manual, Chapter 0530, "Nuclear Criticality Safety," provides guidelines for criticality safety in experimental facilities utilizing fissionable material. In addition, Chapter 0531, "Safety of Non-Reactor Nuclear Facilities" may be applicable to some portions of experimental facilities that are not associated with critical assembly 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 Critical Assembly Core (see Section 4.1), Reactivity Control System (see Section 4.2), Fluid Fuel System (see Chapter 5), 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 consist of critical assemblies which are assembled to perform experiments for the purpose of studying the neutron behavior within the critical assembly, and any ex-core experiments performed in conjunction with the operation of the critical assemblies.

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. If a failure of experimental facility mechanical fluid systems or portions of them may result in potential off-site doses in excess of 0.5 rem, they should be a minimum of Safety Class 3.

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

C. Experimental facilities may be designed to non-nuclear standards if their failure does not result in consequences in excess of the guidelines of Chapter 15.

10.4.2 Guidelines for the Design of Experimental Facilities

A. The design criteria of N405-1975, "Safety Guide for the Performance of Critical Experiments" should be considered in the design of experimental facilities.

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

C. Potential reactivity effects should be within the range for which the reactivity control system (see Section 4.2) was designed. Experimental facility design should consider the potential reactivity effects of the following:

1. Reactor building structures, components, and systems (e.g., moderating effects of concrete structures, piping systems).
2. Experimental facility equipment and apparatus employed in the experiment.
3. Experimental facility equipment and apparatus associated with other experiments in the same reactor building.
4. The concurrent operation of more than one critical assembly in the same reactor building.

Items such as these have the potential to alter reactivity control element worths, modify critical dimensions and shorten neutron lifetimes.

D. The effects of experimental equipment and apparatus on flux measuring equipment should be considered when establishing protection or instrumentation and control system (see Chapter 7) trip or control points.

E. The instrumentation and controls associated with the experimental facilities should include appropriate interlocks if operation of the experimental facility can create conditions that may adversely affect the safety of the critical assembly. If the interlock circuits interface with the Protection System, the guidelines in Section 7.1 should be considered in their design.

F. Experimental facilities should be designed to provide adequate protection against radiation from streaming from the critical assembly core, and scattering from ex-core experiments.

G. Experimental facilities should be designed to prevent or minimize the spread of radioactive contamination. Devices used to prevent the spread of contamination may be part of the experimental facility or part of the critical facility's systems or components (e.g., glove boxes, ventilation systems, leakage catch basins or sumps).

H. Catch basins, sumps, and collection systems which are part of the experimental facilities fluid systems should be designed to maintain sub-critical configurations (e.g., raschig rings, geometry) during normal or postulated accident conditions.

I. Mechanical portions of experimental facilities should consider the design guidelines of Section 3.9.

J. 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 critical facility technical specifications.

2. The proposed experiment should not present unreviewed safety questions.

3. The proposed experiment should not present undue hazards to health and safety or critical facility.

10.4.3 Maintenance, Testing, and Inspection

A. Experimental facility components that form part of the critical assembly core or fluid fuel system boundary should meet the maintenance, testing, and inspection requirements specified in Sections 4.1 and 5.1, respectively.

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

C. Actual experimental equipment should be closely scrutinized to ensure that it conforms with the intent of the experimental plan.

10.4.4 Interfaces With Other Systems

Experimental facilities may interface with the following other facility systems.

- A. Critical Assembly Core
- B. Reactivity Control System
- C. Fluid Fuel System
- D. Reactor Building
- E. Protection System
- F. Non-Safety I&C System

G. Non-Class IE Electric Power System

H. Radioactive Waste System

I. 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 provisions 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 N16.3-1975, "Safety in Conducting Subcritical Neutron Multiplication Measurements in Situ"

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 are distributed only on 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.
3. Davey, W. G. and Redman, W. C., "Techniques in Fast Reactor Critical Experiments," Gordon and Breach, 1970

B. Pending Industry Standards and Codes

1. ANS 15.13, Proposed, "Allowed Core Alterations for Research Reactors"
2. ANSI N16.6 (ANS 8.8), Draft, "Criticality Safety Limits for Special Applications"

10.8 TOPICS FOR FUTURE STANDARD 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 piping, valves, sumps, tanks, drains, evaporators, and process equipment necessary to collect, store, sample and treat waste radioactive 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. Containment/Confinement Cleanup System
2. ESF Ventilation System (Filters and Demister Drains)
3. Fuel Handling and Storage System
4. Evaporator Residues
5. Contaminated Components, or Equipment Used for Maintenance, Repair, Sampling or Testing
6. Cleaning Solvents and Rags
7. Reactor Building Sumps
8. Waste From Radiochemical Analysis
9. Experimental Facilities

B. The radioactive waste system should be designed such that the concentration of radioactive material in routine releases from radioactive waste systems and the exposures resulting from these releases do not exceed the guidelines in the ERDA Manual, Chapters 0524 (see Chapter 12). Releases during accident situations should be within the guidelines of Chapter 15.

C. Portions of the radioactive waste system connected to the critical assembly fluid fuel system boundary should include suitable isolation valves if required to ensure the integrity of a critical assembly fluid system (see Chapter 5).

D. Radioactive waste storage tanks should be designed such that the failure of any one tank, containing the maximum amount of radioactive material,

will not result in exposures to radioactivity in excess of the dose guidelines for the applicable facility operating conditions (see Chapter 15).

E. Waste system catch basins, sumps, tanks, and collection systems should include design features (e.g., raschig rings, geometry) to ensure sub-critical configurations during normal and postulated accident conditions.

F. Monitoring of radioactive waste system effluents should be provided as described in Chapter 12.

G. Radioactive waste system components should be protected against the effects of severe natural and man-made phenomena (see Chapter 3).

H. Mechanical portions of the radioactive waste system should consider the design guidelines of Section 3.9.

I. The capability for reducing the volume of radioactive waste should be provided where practicable.

11.4.3 Maintenance, Testing, and Inspection

A. Safety class fluid portions of the radioactive waste system should be designed to facilitate inservice inspection.

B. 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. Fluid Fuel System
- B. Reactor Building
- C. ESF Ventilation System
- D. Non-Safety I&C System
- E. Non-Class IE Electric Power System
- F. Fuel Handling and Storage System
- G. Fire Protection System
- H. 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 50.35a, "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 3.10, "Liquid Waste Treatment System Design Guide for Plutonium Processing and Fuel Fabrication Plants"

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 are distributed only on 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," *ibid*.
2. WASH-1202, "Plan for the Management of AEC Generated Radioactive Wastes"

B. Pending Industry Standards and Codes

1. ANSI N498 (ANS 40.12), Proposed, "General Criteria for Waste Management"
2. ANSI N525 (ANS 40.11), Proposed, "Definition of Radioactive Waste Categories"
3. ANSI N647 (ANS 15.12), Proposed, "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents"

11.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 12

RADIATION PROTECTION

12.1 SCOPE

Requirements for radiation protection are specified in the following 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 exposures and off-site exposures to radiation and radioactive material as low as reasonably achievable.

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 shall be maintained as low as reasonably achievable 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 as low as reasonably achievable 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. Methods of protection to be considered in facility design include the following:

1. Permanent or Temporary Shielding
2. Ventilation Systems (See Section 6.2)
3. Remote Handling Provisions
4. Area Radiation Zones and Occupancy Restrictions
5. Fixed or Portable Monitoring and Alarm Systems
6. Equipment design to minimize the accumulation of activated corrosion products on internal surfaces of equipment

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 (may actuate a containment/confinement isolation system and/or an ESF ventilation system).
2. Radioactive Waste System Areas
3. Fuel Handling and Storage Areas

4. The facility environment, including gaseous and liquid effluent release paths.

5. 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. Fluid Fuel System
- B. Reactor Building
- C. ESF Ventilation System
- D. Protection System
- E. Class IE Electric Power System
- F. Fuel Handling and Storage System
- G. Experimental Facilities
- H. Radioactive Waste Systems

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

h. ANSI N101.6-1972, "Concrete Radiation Shields"

i. ANSI N237-1975, "Source Term Specification"

j. ANSI N319-1976, "Personnel Neutron Dosimeters"

k. ANSI N322-1977, "Inspection and Test Specification for Direct and Indirect Reading Quartz Fiber Pocket Dosimeter"

1. ANSI N405 (ANS 1), "Safety Guide for the Performance of Critical Experiments"

m. ANSI N512-1974, "Protective Coatings (paints) for the Nuclear Industry"

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

o. 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 Dosimetry"

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

j. ANSI Z88.2-1969, "Procedures for Respiratory Protection"

k. ANSI/ANS 6.1.1-1977 (N666), "Neutron and Gamma Ray Flux-to-Dose Rate Factors"

1. 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, "Personnel Neutron Dosimeters"

2. Radiation Protection System Administration

- a. USNRC Regulatory Guide 1.8, "Personnel Selection and Training"
- b. USNRC Regulatory Guide 1.16, "Reporting of Operating Information--Appendix A, Technical Specifications"
- c. USNRC Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Material in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants:
- d. USNRC Regulatory Guide 1.109, "Calculation of Annual Dose to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I"
- e. USNRC Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors"
- f. USNRC Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors"

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

h. USNRC Regulatory Guide 2.1, "Shield Test Program for Evaluation of Installed Biological Shielding in Research and Training Reactors"

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

j. USNRC Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants"

k. USNRC Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants"

l. USNRC Regulatory Guide 8.1, "Radiation Symbol"

m. USNRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"

n. USNRC Regulatory Guide 8.5, "Immediate Evacuation Signal"

o. USNRC Regulatory Guide 8.7, "Occupational Radiation Exposure Records System"

p. USNRC Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Reasonably Achievable"

q. USNRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is 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 are distributed only on a limited basis as preliminary drafts are listed for information only.

A. Background Information

1. USNRC Standard Review Plan 12.1, "Assuring That Occupational Radiation Exposures are as Low as Reasonably Achievable," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 12.1, "Radiation Sources," ibid.

3. USNRC Standard Review Plan 12.3, "Radiation Protection Design Features," *ibid.*
4. USNRC Standard Review Plan 12.4, "Dose Assessment," *ibid.*
5. USNRC Standard Review Plan 12.5, "Health Physics Program," *ibid.*
6. USNRC Standard Review Plan 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems," *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. WASH-1287, "Manual of Respiratory Protection Against Airborne Radioactive Material," USAEC, 1974
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"
19. NCRP Report No. 8, "Control and Removal of Radioactive Contamination in Laboratories," National Council on Radiation Protection and Measurement

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"

28. "A Guide to Good Practices at Plutonium Facilities," BNWL-2086, September 1977

B. Pending Industry Standards and Codes

1. ANSI N17.1, Proposed, "Location and Range of Measurement of Continuous Area and Process Radiation Monitoring System for Light Water Reactors"

2. ANSI 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, Draft, "Methods for Inferring Environmental Doses"

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

7. ANSI N403, Proposed, "Shield Material Concrete"

8. ANSI N404, Proposed, "Glossary of Terms in Shielding and Dosimetry"

9. ANSI N410, Proposed, "Neutron and Gamma Ray Cross Sections for Nuclear Radiation Protection Calculations"

10. ANSI N647, Proposed, "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents"

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

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

12.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

The following topical listing identifies areas in which the development of additional nuclear standards may result in a consistent approach to critical facility design and modification.

A. Guidelines for providing protection of personnel at DOE reactor facilities against hazards that may be created by reactor system and/or experiment operational anomalies or maintenance activities (e.g., radiation and/or airborne radioactivity hazards to operators and/or experimenters in the reactor building.)

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 should be considered 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 50.59, "Changes, Tests, and Experiments"
- D. 10 CFR 50, Appendix E, "Emergency Plans for Production and Utilization Facilities"
- E. 10 CFR 55, "Operators' License"
- F. 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 N399-1974, "Records and Reports for Research Reactors"
6. ANSI N401-1974, "Review of Experiments for Research Reactors"
7. ANSI N405-1975, "Safety Guide for the Performance of Critical Experiments"
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 1.134, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses"
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 Material Outside Reactors"
4. USNRC Regulatory Guide 5.7, "Control of Personnel Access to Protected Areas, Vital Areas and Material Access Areas"

5. USNRC Regulatory Guide 5.29, "Nuclear Material Control System for Nuclear Power Plants"

6. USNRC Regulatory Guide 5.51, "Management Review of Nuclear Material Control and Accounting System"

7. USNRC Regulatory Guide 5.57, "Shipping and Receiving Control of Special Nuclear Material"

8. USNRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"

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

A. Background Information

1. "A Guide to Good Practices at Plutonium Facilities," BNWL-2086, September 1977

2. "Safe Operation of Critical Assemblies and Research Reactors, 1971 Edition," IAEA Safety Series No. 35, International Atomic Energy Agency, Vienna

3. "Manual for the Operation of Research Reactors," IAEA Technical Report Series No. 37, International Atomic Energy Agency, Vienna, 1964

4. "Planning for the Handling of Radiation Accidents," IAEA Safety Series No. 32, International Atomic Energy Agency, Vienna, 1969

5. EPA-520/1-75-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," USEPA September 1975

B. Pending Industry Standards and Codes

1. ANSI N17.2, Proposed, "Emergency Planning for Research Reactors"

2. ANSI N682, Proposed, "Facilities for Medical Care On and Off Plant Site"

3. ANSI N684, Proposed, "Emergency Control Centers On and Off Plant Site"

4. ANSI N685, Proposed, "Guidance on Conducting Emergency Preparedness Drills"

5. ANSI N700, Proposal, "Standard for Physical Security of 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 N405-1975 (ANS 1), "Safety Guide for the Performance of Critical Experiments"
- 2. IEEE 336-1971, "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"

B. RDT Standards

None

C. Guides

- 1. USNRC Regulatory Guide 1.41, "Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments"

2. USNRC Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems"

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

A. Background Information

1. ASME B&PV Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"

2. IAEA Safety Guide SG-04, Draft, "Commissioning Procedures"

B. Pending Industry Standards and Codes

1. ANSI N267, Proposed, "Requirements for Preoperational and Startup Testing"

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 a critical facility.

15.2 GUIDELINES FOR ACCIDENT ANALYSIS

A. A critical 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 (e.g., a rod withdrawal accident may be accommodated within the consequence guidelines for an emergency operating condition). With the addition of a single failure (discussed in Section 15.3), the critical 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, or it may be necessary to design the system to a final safety class that is higher than the minimum safety class recommended by this Design Guide in order to improve system quality and reliability.

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, and from Plutonium 75 rem dose to the lung, and 150 rem dose to the bone).

TABLE 15.2-1

SPECTRUM OF POSTULATED ACCIDENTS TO BE CONSIDERED IN DESIGN

SOLID FUELED CRITICAL 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. Reactivity control system maloperation (system malfunction, operator error or incapacitation).
3. Inadvertent loading and operation of a fuel assembly in an improper position.
4. Spectrum of accidents in which a reactivity control assembly is withdrawn from the core (including improper location of neutron source or inadvertent movement of source).
5. Spectrum of accidents in which a reactivity control assembly inadvertently drops into the core.
6. Spectrum of reactivity accidents due to operation or failure of experimental facilities.
7. Reactivity changes due to human body moderation, experiment location, or other effects.
8. Reactivity accident in which moderator material may be introduced due to structural failures.

TABLE 15.2-1 (Continued)

B. Radioactive Release from a Subsystem or Component
<ol style="list-style-type: none"> 1. Radioactive liquid waste system leak or failure. 2. Postulated radioactive releases due to liquid tank failures. 3. Fuel handling accidents. 4. Experimental facility or specimen failures. 5. Fires involving plutonium metals. 6. Contamination from bare plutonium.
C. Anticipated Transients Without Scram
<ol style="list-style-type: none"> 1. Inadvertent insertion of positive reactivity by the reactivity control system. 2. Loss of AC power.

TABLE 15.2-1 (Continued)

FLUID FUELED CRITICAL FACILITY

D. Reactivity and Power Distribution Anomalies
<ol style="list-style-type: none"> 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 reactivity control assembly or temporary control device removal error during refueling. 2. Reactivity control system maloperation (system malfunction, operator error or incapacitation). 3. Startup of an inactive fuel fluid loop. 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 core (including improper location of neutron source or inadvertent movement of source). 6. Spectrum of accidents in which a reactivity control assembly inadvertently drops into the core. 7. Spectrum of reactivity insertion accidents due to operation or failure of experimental facilities. 8. Reactivity changes due to flow rates (effective delayed neutron fraction) in fluid fuel systems. 9. Reactivity changes due to human body moderation, experiment location, or other effects.
E. Radioactive Release from a Subsystem or Component
<ol style="list-style-type: none"> 1. Radioactive gas waste system leak or failure. 2. Radioactive liquid waste system leak or failure.

TABLE 15.2-1 (Continued)

<ul style="list-style-type: none"> 3. Postulated radioactive releases due to liquid tank failures. 4. Fuel handling accidents. 5. Experimental facility or specimen failures.
F. Decrease in Critical Assembly Fluid Fuel System Flow Rate
<ul style="list-style-type: none"> 1. Single and multiple fluid pump trips. 2. Fluid pump shaft seizure. 3. Fluid pump shaft break. 4. Fluid system flow blockage due to failure of in-core experiment facility.
G. Decrease in Critical Assembly Fluid Inventory
<ul style="list-style-type: none"> 1. Break in instrument line or other small line connected to the solution system boundary. 2. Loss-of-fuel accidents resulting from the spectrum of postulated piping breaks of the reactor fluid boundary including breaks in experimental facilities that penetrate the fluid system boundary.
H. Increase in Critical Assembly Fluid Inventory
<ul style="list-style-type: none"> 1. Chemical and volume control system or makeup fluid system malfunction (or operator error) that increases fluid inventory.
I. Anticipated Transients Without Scram
<ul style="list-style-type: none"> 1. Inadvertent control rod withdrawal. 2. Loss of AC 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, SHUTDOWN, ASSEMBLY OF CORE	ALARA** ($< \text{ERDAM}$) (0524)	ALARA** ($< \text{ERDAM}$) (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** ($< \text{ERDAM}$) (0524)	ALARA** ($< \text{ERDAM}$) (0524)
EMERGENCY	INFREQUENT INCIDENTS OR OCCURRENCES WHICH MAY OCCUR DURING THE LIFETIME OF A FACILITY	$10^{-1} > F \geq 10^{-3}$	REACTIVITY EXCURSION	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	$10^{-3} > F \geq 10^{-7}$	NUCLEAR BURST	$\leq 10\text{CFR100}$ AND FROM PLUTONIUM: $\leq 75 \text{ rem LUNG}$ $\leq 150 \text{ rem BONE}$	$\leq 25 \text{ rem WHOLE BODY}$ AND $\leq 300 \text{ rem THYROID AND FROM PLUTONIUM}$: $\leq 75 \text{ rem LUNG}$ $\leq 150 \text{ rem BONE}$

*THE TABULATED FREQUENCY RANGES FOR EACH OPERATING CONDITION ARE ILLUSTRATIVE EXAMPLES ONLY. THEY ARE BASED ON THE WRITTEN DESCRIPTION IN THE TABLE, FOR A FACILITY WITH AN EXPECTED LIFETIME IN THE 20-40 YEAR RANGE.

**AS LOW AS REASONABLY ACHIEVABLE.

TABLE 15.2-2 (CONTINUED)

OPERATING CONDITION	GENERAL GUIDELINES	FUEL DESIGN GUIDELINES	MECHANICAL SYSTEM DESIGN GUIDELINES
NORMAL	NO PROTECTION SYSTEM ACTION REQUIRED	NO FUEL OR CLAD- DING DAMAGE (SEE SECTION 4.1)	EQUIPMENT LOADING ALLOWABLE STRESS, AND QUALIFICATION (SEE SECTION 3.9, 3.10, AND 3.11)
UPSET	THE FACILITY SHOULD BE CAPABLE OF A RAPID RETURN TO OPERATION WITHOUT EXTENSIVE CORRECTIVE ACTION OR REPAIR		
EMERGENCY	FACILITY SHOULD BE CAPABLE OF RETURN- ING TO OPERATION FOLLOWING CORRECTIVE ACTION OR REPAIR OF DAMAGE	ANY FUEL OR CLAD- DING 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 in 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 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 (75 rem lung dose, 150 rem bone dose from plutonium).

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 (75 rem lung dose, 150 rem bone dose from plutonium).

3. Dose Equivalent

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 comparable to the corresponding whole body dose or thyroid dose guidelines.

C. The minimum safety class to which components, systems, and structures should be designed is determined based on 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. Due to specific features of an individual Category I critical facility, analysis of the design basis accidents may indicate that some consequences are not within the guidelines of Table 15.2-2. Therefore, some system design changes may be necessary to ensure that a final safety class equivalent to the recommended minimum safety class will result in consequences within the guidelines of Table 15.2-2. Alternatively, it may be necessary to increase

the quality and reliability of the system by designing to a final safety class that is higher than the minimum safety class recommended by this Design Guide.

D. The spectrum of Design Basis Accidents to be considered in the design of critical facilities are 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. Probability guidelines applicable to the determination of the spectrum of severe natural phenomena to be considered in critical facility design are described in Table 15.2-2 and in E. below.

Based on the likelihood of occurrence during the life of a critical facility, design basis accidents may be classed as upset, emergency or faulted operating condition.

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. Based on 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 critical facility based on the expected facility operating lifetime and the nature and consequences of accidents postulated for the facility.

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 the low probability events under consideration, 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.

F. The accident analyses should make conservative assumptions regarding the values of all parameters which may have an effect on the calculated consequences of postulated accidents. Some particular conservatisms to be included in the accident analysis are listed below:

1. The short term diffusion estimates used for calculating the radiological consequences of postulated accidents should consider the guidance in USNRC 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 critical assembly power level assumed in the accident analysis should be 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 which results in the loss of capability of a component to perform its intended safety functions. 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 critical facilities, systems such as the protection 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. 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 I reactors may have different system requirements than commercial power reactors. Some Category I 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, pumps). 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 pump or a standby diesel generator to start 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 systems pressure boundaries, flow blockage in fluid systems, and electric short or open circuits.

C. Application of the Single Failure Criterion

Critical 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 rod withdrawal).
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.
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 battery bank to operate or a single failure of any other type of emergency 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 critical 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 means of 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 purpose 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 electrical single failure, or
2. any active fluid system failure (e.g., failure of a valve to close, or a pump to start).

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 such as a containment/confinement isolation system.

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 electrical single failure, or
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 such as a confinement ventilation system.

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 critical facility:

A. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

B. 10 CFR 100, "Reactor Site Criteria"

15.5 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered when performing or evaluating the analysis of postulated accidents at a critical facility:

A. Industry Standards and Codes

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

2. IEEE 379-1977 (ANSI N41.2), "Application of the Single Failure Criterion to Nuclear Power Generating Station Class IE Systems"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems"

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

15.6 SUPPLEMENTARY INFORMATION

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

A. Background Information

1. USNRC Standard Review Plan 2.3.4, "Short Term Diffusion Estimates," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 15.7.2, "Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)," *ibid.*

3. USNRC Standard Review Plan 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures," *ibid.*

4. USNRC Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," *ibid.*

5. USNRC Standard Review Plan 15.7.5, "Spent Fuel Cask Drop Accidents," *ibid.*

6. Branch Technical Position EICSB 18, "Application of the Single Failure Criterion to manually controlled Electrically Operated Valves," SRP Appendix 7A) *ibid.*

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

8. Slade, D. H. (Editor), "Meteorology and Atomic Energy 1968," TID-24190, USAEC, 1968

9. Briggs, G. A., "Plume Rise," TID-25075, USAEC, 1969

10. Sesonske, Alexander, "Nuclear Power Plant Design Analysis," TID-26241, USAEC, 1973

11. Turner, D. G., "Workbook on Atmospheric Dispersion Estimates," AP-26, Revised, "USEPA Office of Air Programs, 1970

12. 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.

13. Winton, M. L., "Computer Codes for Analyzing Nuclear Accidents," Nuclear Safety, Vol. 15, No. 5, September - October 1974

B. Pending Industry Standards and Codes

None

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 consistent approach to critical facility design and modification:

A. Guidelines for a standard approach to calculating doses from postulated critical facility accidents.

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 critical 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 critical facilities.

A. 10 CFR 50.36, "Technical Specifications"

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

A. Background Information

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

2. USNRC Standard Review Plan 2.4.14, "Technical Specifications and Emergency Operation Requirements (Hydrology)", NUREG 75/087, September 1975

3. IAEA Safety Guide SG-03, Draft, "Safety Guide on Operational Limits and Conditions"

B. Pending Industry Standards and Codes

None

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 critical facility.

Generally, critical facilities are 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 critical 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 rather than 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 critical facility design require: structures, components, and systems designed to the ASME Code, (see Section 3.2), a containment building, (see Section 6.1); or Class IE Electric Power Systems, (see Section 8.1), the designer should refer to the Design Guide for Category II 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. 10 CFR 50.55a, "Codes and Standards"

B. 10 CFR 50, Appendix A, General Design Criteria 1 "Quality Standards and Records"

C. 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. ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities"
2. ANSI N45.2.10-1973, "Quality Assurance Terms and Definitions"
3. ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"
4. ANSI N399-1974, "Records and Reports for Research Reactors"
5. ANSI N402-1976, "Quality Assurance Program Requirements for Research Reactors"
6. ANSI Z201.1-1973 (ASTM E380-72), "Metric Practice Guide"
7. IEEE 268-1975, "Standard for Metric Practice"
8. IEEE 336 (ANSI N45.2.4), "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations"

B. RDT Standards

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

1. F 2-2T, "Quality Assurance Program Requirements"
2. F 2-4T, "Quality Verification Program Requirements"
3. F 2-7T, "Acceptance Sampling Plans"
4. F 2-9T, "Reliability Assurance"
5. F 3-2T, "Calibration Program Requirements"
6. F 5-1T, "Cleaning and Cleanliness Requirements for Nuclear Components"

C. Guides

1. USNRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)"
2. USNRC Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"

3. USNRC Regulatory Guide 1.74, "Quality Assurance Terms and Definitions"

4. USNRC Regulatory Guide 2.5, "Quality Assurance Requirements for Research Reactors"

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

A. Background Information

1. WASH 1283, Revision 1, "Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," (Grey Book), USAEC, May 1974

2. WASH 1284, "Guidance on Quality Assurance Requirements During the Operational Phase of Nuclear Power Plants," (Orange Book), USAEC, May 1974

3. WASH 1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," (Green Book), USAEC, October 1973

4. "Matrix of Nuclear Quality Assurance Program Requirements," American Society for Quality Control, April 1973

5. "Quality Systems in the Nuclear Industry," STP 616, American Society for Testing and Materials"

B. Pending Industry Standards and Codes

1. 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"

2. 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"

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

17.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 18

Distribution List

Comment Sheets

DISTRIBUTION

U.S. Department of Energy

Manager
Albuquerque Operations Office

Manager
Chicago Operations Office

Manager
Idaho Operations Office

Manager
Oak Ridge Operations Office

Manager
Richland Operations Office

Manager
San Francisco Operations Office

Manager
Savannah River Operations Office

Manager
Nevada Operations Office

DATE:

TO: Reactor Standards Group - Bldg. 703
Brookhaven National Laboratory
Upton, L.I., N.Y. 11973

FROM:

SUBJECT: "Design Guide For Category I Reactors Critical Facilities"