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DESIGN GUIDE

FOR

CATEGORY III REACTORS

POOL TYPE REACTORS

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

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DESIGN GUIDE FOR CATEGORY III REACTORS POOL TYPE REACTORS

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FORWORD

The Department of Energy (DOE) owns and operates a wide variety of nuclear reactors and critical facilities as part of a comprehensive program which supports the development of advanced energy systems. As required by the ERDA Manual, Chapter 0540, DOE-owned reactors and critical facilities are sited, designed, constructed, modified, operated, maintained and decommissioned in a manner that gives adequate considerations 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 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 power levels, lower fission product inventories, remote siting, and reduced dependence on the operation of systems required to assure safe reactor shutdown and cooling and to assure acceptable consequences of postulated accidents. As a result, simplified and/or unique DOE design and performance requirements achieve the same defense-in-depth characteristics as the larger, more complicated commercial power reactors.

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

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 system in order to adequately protect health and safety.

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

Table A-1. ERDA Reactor Categories.

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

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

Table A-2. DOE Category III (Pool-Type) Reactors

Category III Reactor Subgroup ⁽¹⁾	ERDA Reactor	Actual Maximum Steady- State Thermal Power Level	Reactor Location	Notes
1 MW < P ≤ 10 MW	BSR	1 MW	ORNL	-
P ≤ 1 MW	TRIGA-ACPR HEDL TRIGA NRAD TRIGA CFRMF ETRC PCA ARMF ATRC	600 KW 250 KW 250 KW 100 KW 50 KW 10 KW 10 KW 5 KW	Sandia HEDL INEL INEL INEL ORNL INEL INEL	Pulsing Capability ⁽¹⁾ - ex-PRNC TRIGA - (3) (3) - (3)

(1) P = Maximum steady state thermal power level of Category III reactor.

(2) Also see Design Guide for Category V Reactors for guidance on transient-related aspects of reactor design.

(3) Also see Design Guide for Category I Reactors for alternative critical facility design guidelines.

(q)A

the general topical areas described above, the Design Guide recommends industry and NRC standards, codes and guides and the applicable portions of 10 CFR (including the General Design Criteria in 10 CFR 50, Appendix A) to be considered by the DOE reactor facility contractor.

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

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

A program will be developed to periodically update and revise the Design Guide as necessary to reflect future standards, revisions and developments and to incorporate changes recommended by the Design Guide users. In this manner, the Design Guide will remain a timely standard reference for the designer and reviewer, alike.

CHAPTER 1

INTRODUCTION

1.1 INTRODUCTION

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 health and safety factors. 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 reactors be in accordance with generally uniform standards, guides, and codes which are comparable to those applied to similar reactors licensed by the Nuclear Regulatory Commission (NRC).

1.3 SCOPE

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

1.3.1 General Category III Reactor Characteristics

This guide is specifically applicable to pool-type reactors. These reactors have the following general characteristics:

A. The reactor operates in an open, unpressurized pool, with cooling generally by natural convection of light water primary coolant, although operations at high power (e.g., >1 MW) may require supplementary forced circulation.

B. The reactor normally operates at a steady-state power level except during periods of startup, shutdown, and routine power level adjustments (Category III reactors with additional transient or pulse capabilities may use this Design Guide for the steady-state aspects of the facility design, but should additionally consider the guidelines specified in the Design Guide for Category V reactors which presents additional guidelines for pool type reactors with transient capabilities.

1.3.2 Category III Reactor Subgroups

The Category III reactors are divided into two broad subgroups based on the maximum steady-state thermal power of the reactor. Within a subgroup, the reactors have generally similar system characteristics. The power ranges which define the subgroups are:

A. Power $\leq$ 1 MW

B. 1 MW < Power $\leq$ 10 MW

The first subgroup of Category III reactors ($\leq$ 1 MW) includes the pool type reactors that generally do not require active systems for reactor heat removal. Natural convection provides adequate heat removal during normal operations. No residual heat removal system or emergency core cooling system is usually required for shutdown heat removal or core cooling following a Loss of Coolant Accident (LOCA), respectively.

The second subgroup of Category III reactors (1 MW < Power $\leq$ 10 MW) includes the pool type reactors that generally require forced circulation for high power (>1 MW) operation and may require an emergency core cooling system to provide adequate core cooling following a LOCA.

There are no DOE-owned Category III reactors operating at greater than 10 MW maximum steady-state thermal power, therefore, additional subgroups of Category III reactors in this higher power range are not necessary.

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 safely shutdown 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 their potentially large inventories of radioactive material available for release, power reactors provide multiple physical barriers to restrict the accidental release of radioactivity to the environment. These barriers generally include the fuel cladding, the reactor coolant system pressure boundary, and a containment. Due to the great differences between Category III DOE-owned reactors and power reactors, it is possible to simplify the design of the DOE reactor facility while still adequately protecting health and safety. In general, Category III reactors present much less risk to health and safety than do power reactors because of factors such as much lower source terms, remote siting, and reduced dependence on the operation of systems required to assure safe reactor shutdown and cooling and/or to prevent an uncontrolled release of radioactivity to the environment. As a result, Category III DOE reactor facilities require fewer physical barriers to the release of radioactivity to the environment than power reactors in order to remain within the applicable off-site and on-site radiological consequence guidelines (see Table 1.4-1). For example, Category III DOE reactor facilities generally utilize a reactor building that does not control the release of radioactivity to the environment as a containment would for commercial power reactors. Defense-in-depth is provided at these DOE facilities by the physical barrier represented by the fuel cladding. In addition, inherent safety features such as reduced source terms (due largely

Table 1.4-1. Summary of the General Safety Philosophy for Facility Operation in Terms of Frequency of Occurrence and Radiological Consequence Guidelines

Operating Condition	Description	Typical Design Basis Events	Radiological Consequence Guidelines	
			Off-Site	On-Site
Normal	Events that are planned to occur regularly in the course of plant operation	Startup, shut-down, refueling	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	ALARA* ≤ ERDAM
Emergency	Infrequent incidents or occurrences which may occur during the lifetime of a facility	Rod withdrawal accident	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	Loss of coolant accident	≤ 10CFR100	≤ 25 rem whole body and ≤ 300 rem thyroid

* As low as reasonably achievable.

to lower operating power levels), enhanced dispersion of released radioactivity (due to remote siting) and the ease of surveillance of the reactor and associated systems (due to the open reactor pool) provide the required reasonable assurance that the DOE-owned reactor 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 frequency of occurrence and radiological consequence guidelines corresponding to each facility operating condition (normal, upset, emergency and faulted). The facility operating conditions are discussed in detail in Section 3.9 and in Chapter 15. The radiological consequence guidelines in Table 1.4-1 are to be interpreted as minimum guidelines to protect health and safety. Each DOE contractor has the additional responsibility for maintaining radiation exposures As Low As Reasonably Achievable (ALARA) (see Chapter 12).

A. Safety Class and Safety Function

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

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

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

B. Designation of Safety Class Structures, Components and Systems

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

1. Minimum Safety Class

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

2. Final Safety Class

Based on the accident analysis performed during the Preliminary Safety Analysis Report (PSAR) and Final Safety Analysis Report (FSAR) stages of facility review, the adequacy of the minimum safety class recommended by the Design Guide can be determined. Due to specific features of an individual Category III reactor facility, analysis of some design basis accidents may indicate that the on-site and/or off-site radiological consequences exceed the guidelines in Table 1.4-1. Therefore, some system design changes may be necessary to ensure that a final safety class, equivalent to the recommended minimum safety class, will result in consequences within the guidelines in Table 1.4-1. Alternatively, it may be necessary to increase the quality and reliability of the system by designing it to a final safety class that is higher than the minimum safety class recommended by this Design Guide.

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

1.42 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 reactor facilities and to the modification of significant portions of existing facilities. For each system, the Design Guide provides guidance for determining if the system is required at a particular facility and, if the system is required, the Design Guide recommends major safety concepts and design philosophies and the codes, standards, and guides to be considered in system design and/or modification.

Some existing industry standards and codes and USNRC Regulatory Guides, although written primarily for commercial power reactors, are also generally applicable to the design of Category III reactors. Where necessary, design guidelines provide interpretations of the standards, codes, and guides in order to facilitate their application to Category III reactors. When existing standards, codes, and guides do not provide adequate guidance, the design guidelines presented in this Design Guide serve as interim recommendations pending the development or approval of appropriate new standards, codes, and guides.

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

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

1.4.3 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 reactor facility or modifying an existing reactor facility. As illustrated in Figure 1.4-1, the design of any system should consider the guidance provided in Chapter 3 and in the appropriate chapter or section which presents the detailed system design guidelines.

Table 1.4-2 lists the systems that are typically associated with Category III reactors and also references the Design Guide chapter or section that provides the detailed system design information.

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

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

Figure 1.4-1. Design Guide Organization.

Table 1.4-2. Typical Systems for Category III Reactors⁽¹⁾

(Power $\leq$ 1 MW)

System	Design Guide Section
Reactor Core Assembly	4.1
Reactivity Control System	4.2
Reactor Coolant System ⁽²⁾	5.1
Reactor Building ⁽³⁾	6.2
Protection System	7.1
Non-Safety I&C System	7.2
Non-Class IE Electric System	8.2
Fuel Handling and Storage System	9.2
Purification and Make-Up Water System	9.3
Fire Protection System	9.4
Experiment Facilities	10
Radioactive Waste System	11
Radiation Protection System	12

- (1) The systems listing should not be interpreted as a mandatory minimum systems requirement, but rather as a listing of the systems that would be found at a hypothetical "typical facility" and the corresponding Design Guide section or chapter references.
- (2) Generally, the reactor is cooled by natural circulation of the reactor coolant in the pool. Reactor coolant loop piping external to the pool is not generally required.
- (3) Generally, this reactor building need not be capable of controlling the release of radioactivity to the environment.

Table 1.4-2. (Continued). Typical Systems for
Category III Reactors⁽¹⁾

(1 MW < Power ≤ 10 MW)

System	Design Guide Section
Reactor Core Assembly	4.1
Reactivity Control System	4.2
Reactor Coolant System ⁽⁴⁾	5.1
Emergency Core Cooling System ⁽⁵⁾	6.1
Reactor Building	6.2
ESF Ventilation System	6.3
Protection System	7.1
Non-Safety I&C System	7.2
Class IE Electric Power System	8.1
Non-Class IE Electric Power System	8.2
Cooling Water System	9.1
Fuel Handling and Storage System	9.2
Purification and Make-Up Water Systems	9.3
Fire Protection System	9.4
Experimental Facilities	10
Radioactive Waste Systems	11
Radiation Protection Systems	12

(4) Includes reactor coolant loop piping external to the pool, if provided.

(5) Suitable Technical Specification limits on reactor power history may permit operation without an ECCS.

E. General Design Guidelines.

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

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

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

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

1.4.4 Maintenance of the Design Guide

The continuing development and revision of standards, codes, and guides that are applicable to Category III reactor facilities will necessitate the periodic updating of the Design Guide in order to ensure that the guidance being provided to the 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 Guide and for providing additional information as necessary to the Design Guide users during the period between revisions. Chapter 18 includes preaddressed comment sheets to facilitate feedback from the Design Guide users. Comments will be reviewed and changes will be made to the Design Guide, as necessary, at its next scheduled revision.

1.5 DEFINITIONS

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

1. ALARA

Pertaining to maintaining exposures of personnel to radiation during routine operation of a reactor facility "As Low As Reasonably Achievable"

(10 CFR 50, Appendix I and USNRC Regulatory Guide 8.8). This term is comparable to ALAP, "As Low As Practical," discussed in ERDAM 0524.

2. Class IE

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

3. 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).

4. Design Basis Accidents

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

5. 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).

6. Loss-of-Coolant-Accidents (LOCA)

Those postulated accidents that result in a loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in the reactor coolant system boundary (10 CFR 50, Appendix A).

7. Off-Site Dose

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

8. On-Site Dose

Potential radiological exposure to persons within the exclusion area boundary.

9. 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 plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for

which those features of the nuclear power plant necessary for continued operation without undue risk to health and safety are designed to remain functional (10 CFR 100, Appendix A).

10. Operating Conditions

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

11. 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 shut down the reactor and maintain it in a safe shutdown, or (b) the capability to prevent or mitigate the consequences of accidents which would result in potential off-site exposures comparable to the guideline exposures of 10 CFR 100. (10 CFR 100, Appendix A, Revised.)

12. Safety Class

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

13. Safety Function

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

14. 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. Operations beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency. (10 CFR 50.36c, Revised.)

15. 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).

16. 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).

17. Uncontrolled Area

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

CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Guidelines for Reactor Siting

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

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

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

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

2.1.2 Applicable Siting Criteria

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

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

B. 10 CFR 100, "Reactor Site Criteria"

2.1.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes

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

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.78, "Assumption for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"

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

3. USNRC Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations"

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

5. USNRC Regulatory Guide 4.11, "Terrestrial Environmental Studies for Nuclear Power Stations"

2.1.4 Supplementary Information

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

B. Pending Industry Standards and Codes

1. ANSI N634 (ANS 2.6), Proposed, "Guidelines for Estimating Present and Future Population Distributions Surrounding Power Reactor Sites"

2. ANSI N635 (ANS 2.12), Draft, "Guidelines for Combining Natural and External Manmade Hazards at Power Reactor Sites"

2.1.5 Topics for Future Standards Development

None

2.2 METEOROLOGY

2.2.1 Guidelines for Reactor Siting

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

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

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

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

D. The long-term diffusion estimates used for calculating the radiological consequence of normal releases should also consider the guidance in TID-24190 regarding the acceptability of models and conservatism.

2.2.2 Applicable Siting Criteria

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

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

B. 10 CFR 100, "Reactor Site Criteria"

2.2.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes

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

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

B. RDT Standards

None

C. Guides

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

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

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

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

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

2.2.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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 meteorology-related bibliographies.

1. USNRC Standard Review Plan 2.3.1, "Regional Climatology," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 2.3.2, "Local Meteorology," *ibid.*

3. USNRC Standard Review Plan 2.3.3, "On-Site Meteorological Measurement Programs," *ibid.*

4. USNRC Standard Review Plan 2.3.4, "Short-Term (Accident) Diffusion Estimates," *ibid.*

5. USNRC Standard Review Plan 2.3.5, "Long-Term (Routine) Diffusion Estimates," *ibid.*

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

7. U.S. Department of Commerce, "Climatic Atlas of the United States," Environmental Data Service, NOAA, June 1968

8. U.S. Department of Commerce, "Local Climatological Data and Comparative Data," Environmental Data Service, NOAA, Published Annually for all First-Order NWS Stations

9. U.S. Department of Commerce, "State Climatological Summary," Environmental Data Service, NOAA, Published Annually by State

10. Holzworth, G.C., "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Office of Air Programs, USEPA, January 1972

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

12. Yanskey, G.R., E.H., Markee, and A.P. Richter, "Climatography 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"

15. IAEA Safety Guide SG-S11B, Draft, "Design Basis Hurricane-Wind and Pressure Field"

B. Pending Industry Standards and Codes

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

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

2.2.5 Topics for Future Standards Development

None

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2.3 HYDROLOGY

2.3.1 Guidelines for Reactor Siting

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

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

B. Category III reactors do not generally require safety class cooling water systems (with the possible exception of an ECCS as described in Section 6.1). Water sources (e.g., from rivers, groundwater, etc.) thus need not be protected against the effects of severe natural phenomena.

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

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

2.3.2 Applicable Siting Criteria

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

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

B. 10 CFR 100, "Reactor Site Criteria"

2.3.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes

1. ANSI N170-1976 (ANS 2.8), "Standards and Determining Design Basis Flooding at Power Reactor Sites"

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

B. RDT Standards

None

C. Guides

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

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

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

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

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

2.3.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry code 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, "Probable Dam Failures (Seismically Induced)," *ibid.*

5. USNRC Standard Review Plan 2.4.5, "Probable Maximum Surge and Seiche Flooding," *ibid.*

6. USNRC Standard Review Plan 2.4.6, "Probable Maximum Tsunami Flooding," *ibid.*

7. USNRC Standard Review Plan 2.4.10, "Flood Protection Requirements," *ibid.*

8. USNRC Standard Review Plan 2.4.12, "Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters," *ibid.*

9. USNRC Standard Review Plan 2.4.13, "Groundwater," *ibid.*

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

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

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

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

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

B. Pending Industry Standards and Codes

1. ANSI N181 (ANS 2.9), Proposed, "Guidelines for Evaluating Ground Water Supply at Nuclear Power Sites"

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

3. ANSI N718 (ANS 2.13), Proposed, "Standard for Evaluating Surface Water Supply for Power Reactor Sites:

4. ANS 2.17, Proposed, "Guidelines for Evaluating Ground Water Transportation of Radionuclides at Nuclear Power Sites:

5. ANS 2.18, Proposed, "Guidelines for Evaluating Surface Water Transport of Radionuclides at Nuclear Power Sites"

2.3.5 Topics for Future Standards Development

None

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2.4 GEOLOGY AND SEISMOLOGY

2.4.1 Guidelines for Reactor Siting

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

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

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

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

2.4.2 Applicable Siting Criteria

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

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

B. 10 CFR 100, "Reactor Site Criteria"

2.4.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes

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

B. RDT Standards

None

C. Guides

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

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

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

2.4.4 Supplementary Information

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

B. Pending Industry Standards and Codes

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

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

2.4.5 Topics for Future Standards Development

None

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 INTRODUCTION TO DESIGN CLASSIFICATION

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

Figure 3.1-1 is a guide to the use of Chapter 3. As illustrated in this figure, the severe natural and man-made phenomena which should be considered in the design of facility systems are those which may prevent safe shutdown and cooling of the reactor or which may cause 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 facility design are also discussed in Chapter 15.

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. Some of these severe natural phenomena, such as hurricanes and floods, can be forecast with sufficient accuracy and timeliness to allow some protective actions to be taken in advance of their most severe effects on a reactor facility. Other severe phenomena, such as earthquakes cannot be accurately forecast and, therefore, may occur with little or no warning.

Unlike power reactors, which may be designed to continue operating during some severe natural phenomena in order to supply electric power, Category III reactors are operated intermittently on a flexible schedule that would permit shutting down the reactor when the facility is threatened by impending severe natural phenomena. At these facilities, some safety class equipment that may be required during reactor operation (e.g., protection system) may not need protection from some severe natural phenomena if a timely reactor shutdown is required in advance of the severe natural phenomena and the systems are then no longer required for safety (e.g., due to the scrammed condition of the reactor). Other safety class systems that are required for maintaining the reactor in a safe

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

Figure 3.1-1. Guide to the Use of Chapter 3 (Continued).

shutdown condition or for preventing an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15 should be protected against the effects of the severe natural phenomena.

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

Category III reactor safety class components, systems, and structures should be designed to withstand severe natural phenomena that cannot be accurately forecast (e.g., earthquakes) to ensure that the system safety functions can be completed when necessary.

Table 3.1-1 summarizes the principle code requirements and design standards that are generally applicable to the design of Category III reactor facility components, systems and structures.

Table 3.1-1. Principal Code and Design Requirements for Component Safety Class 3 Components and for Non-Safety Class (Non-Nuclear) Components

Component	Safety Class 3 (Quality Group C)	Non-Nuclear (Quality Group D)
Pressure vessels	ASME B and PV Code, Section III, Division I, Class 3	ASME B and PV Code, Section VIII, Division I
Pumps	ASME B and PV Code, Section III, Class 3	Manufacturer's Standards
Valves	ASME B and PV Code, Section III, Class 3	ANSI B16.5 and B31.1
Piping	ASME B and PV Code, Section III, Class 3	ANSI B31.1
0 to 15 psig storage tanks	ASME B and PV Code, Section III, Class 3	API-620 or equivalent
Atmospheric storage tanks	ASME B and PV Code, Section III, Class 3	API-650 AWWA-D100, or ANSI B96.1
Heat exchangers	ASME B and PV Code, Section III, Class 3 and TEMA "R"	ASME B and PV Code, Section VIII, Division I and TEMA "C" or "R"

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3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

Components, systems and structures required for assuring safe shutdown and cooling of a reactor and/or for preventing an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15 are designated Seismic Category I and are specifically designed to withstand the effect of the Safe Shutdown Earthquake (SSE). The SSE is based on an evaluation of the maximum earthquake potential of the reactor site and is that earthquake which produces the maximum vibratory ground motion for which safety class structures, components, and systems are designed to remain functional.

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 I 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 Electrical Safety Classification

Electrical components and systems required for assuring safe shutdown and cooling of the reactor and/or for preventing an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15 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 normal industrial design and QA requirements.

Category III reactors do not generally require Class IE electric power if the following conditions are met:

A. The reactor is designed to inherently fail in a safe shutdown condition upon loss of electric power.

B. An active, electric powered system is not required for shutdown cooling of the reactor (including post-LOCA core cooling).

C. An active electric powered system is not required to prevent an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15.

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 Class 1, 2 and 3 corresponding directly to Quality Groups A, B and C. Non-Safety

Class (or non-nuclear) components are in quality group D. USNRC Regulatory Guide 1.26 requires that a system be designed to a safety class if the conservatively calculated potential off-site doses due to failure of the system can exceed 0.5 rem to the whole body. If the conservatively calculated potential off-site doses due to failure of the system do not exceed 0.5 rem, the system may be designed to non-nuclear standards. Table 3.2-1 presents general guidelines for determining the minimum safety class of Category III reactor facility components, systems, and structures. Note that Category III reactors do not require Safety Class 1 components, systems and structures. SC-1 applies to reactor coolant system pressure boundary components of commercial power reactors. The reactor coolant system of Category III reactors is an open, unpressurized system that is not relied upon as a physical barrier to the release of radioactivity to the environment.

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 each subgroup of Category III reactors. Table 3.2-1 is not to be interpreted as a listing of required systems for each subgroup of Category III reactors. It is rather a listing of systems that would be found at a hypothetical "typical facility." Individual facilities may vary substantially from this listing while still adequately protecting health and safety. Due to significant differences between Category III reactors and power reactors, the Table 3.2-1 and 3.2-2 listings do not necessarily correspond to the safety class guidelines that are applicable to power reactors.

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

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

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

Category III reactors generally require few components to be designed to the requirements of the ASME B&PV code. If a particular Category III reactor facility should determine that some systems or components should be designed to the requirements of the ASME B&PV code, that facility should consult the Design Guide for Category II Reactors for the appropriate design guidelines.

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

SAFETY CLASS 1 (Quality Group A)

Category III reactors do not require Safety Class 1 components, systems or structures.

SAFETY CLASS 2 (Quality Group B)

- A. Those systems, the failure of which may result in the radiological consequences of accidents being in excess of the 10CFR100 guidelines.
- B. Those systems, the failure of which would prevent safe shutdown and/or adequate core shutdown heat removal.

SAFETY CLASS 3 (Quality Group C)

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

Note (1) General guidance for determining the final safety class are presented in Section 3.2 and in Chapter 15.

- (2) Components, systems and structures that contain or may contain radioactive material but are not Safety Class 1, 2 or 3 may be designed to Non-Safety Class (Non-Nuclear) standards. The maximum off-site dose resulting from the failure of a Non-Nuclear System should not exceed 0.5 rem to the whole body or the equivalent to any part of the body.

Table 3.2-2. Typical Equipment Classification for Category III Reactors.
(Maximum Power $\leq$ 1 MWt)

System or Component	Design Guide Section	Safety Class	Quality Group	Seismic Category	10CFR50 Appendix B	Principal Design Code or Standard
<u>Power $\leq$ MWt:</u>						
Reactor Core Assembly	4.1	—	—	I	Applies	—
Reactivity Control System	4.2	—	—	I	Applies	—
Reactor Coolant System	5.1	NN ^(a)	D	—	—	API 650, AWWA D-100
Reactor Building	6.2	NN ^(c)	D	—	—	ANSI A89.1 AISC Specifications
Protection System	7.1	—	—	I	Applies	IEEE 279
Non-Safety Instrumentation and Control	7.2	NN	D	—	—	NEMA ICS-1970
Non-IE Electric System	8.2	NN	D	—	—	ANSI C1-1975 NEMA MG 1-1972
Fuel Handling and Storage System	9.2	NN	D	—	—	API 650, AWWA D-100
Purification and Make-up Water System	9.3	NN	D	—	—	ANSI B31.1-1977
Fire Protection System	9.4	NN	D	—	—	NFPA 802-1974
Experimental Facilities	10	(d)	(d)	(d)	(d)	—
Radioactive Waste System	11	NN	D	—	—	ASME B&PV Code Section VIII ANSI B31.1-1977
Radiation Protection System	12	NN	D	—	—	—

3-4(b)

III

Table 3.2-2. (Continued) Typical Equipment Classification for Category III Reactors.
(Maximum Power >1 MWt but $\leq$ 10 MWt)

System or Component	Design Guide Section	Safety Class	Quality Group	Seismic Category	10CFR50 Appendix B	Principal Design Code or Standard
<u>1 MWt < Power $\leq$ MWt</u>						
Reactor Core Assembly	4.2	—	—	I	Applies	—
Reactivity Control System	4.2	—	—	I	Applies	—
Reactor Coolant System	5.1	3	C	I	Applies	API-650, AWWA D-100, ASME B&PV Code, Section III
Emergency Core Cooling System ^(b)	6.1	3	C	I	Applies	ANSI B31.1-1977, ASME BP&V Code, Section III
Reactor Building	6.2	3 ^(c)	C	I	Applies	ANSI A89.1, AISC Specs.
ESF Ventilation System	6.3	3	C	I	Applies	Regulatory Guide 1.52
Protection System	7.1	—	—	I	Applies	IEEE 279
Non-Safety Instrumentation and Control System	7.2	NN	—	—	—	NEMA ICS-1970
Class IE Electric System	8.1	—	—	I	Applies	IEEE 308
Non-Class IE Electric System	8.2	NN	—	—	—	ANSI C1-1975 NEMA MG 1-1972
Cooling Water Systems	9.1	NN	D	—	—	ANSI B31.1-1977
Fuel Handling and Storage System	9.2	NN	D	—	—	API-650, AWWA D-100
Purification and Make-up Water System	9.3	NN	D	—	—	ANSI B31.1-1977
Fire Protection System	9.4	NN	D	—	—	NFPA 802-1974

3-4(c)

III

Table 3.2-2. (Continued) Typical Equipment Classification for Category III Reactors.
(Maximum Power >1 MWt but $\leq$ 10 MWt)

System or Component	Design Guide Section	Safety Class	Quality Group	Seismic Category	10CFR50 Appendix B	Principal Design Code or Standard
<u>1 MWt < Power $\leq$ 10 MWt</u>						
Experimental Facilities	10	(d)	(d)	(d)	(d)	—
Radioactive Waste System	11	NN	D	—	—	ANSI B31.1-1977
Radiation Protection System	12	3	C	I	Applies	—

Notes:

- (a) See guidelines, Table 5.1-1.
- (b) No ECCS required with suitable Technical Specifications to limit reactor power history (see Subsection 6.1).
- (c) See guidelines, Table 6.2-2.
- (d) Experimental facilities may interface with many facility systems, including the reactor core assembly, reactor coolant system boundary, instrumentation, control and electric systems. The experimental facilities should be designed to a safety class, quality group, seismic category and QA standard that is compatible with the design of the interfacing system.

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

Structures, components, and systems not subject to the ASME Boiler and Pressure Vessel Code may be Safety Class 1, 2 or 3 or 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 ensure that mechanical components are designed with sufficient conservatism to ensure the safe shutdown and cooling of the reactor and/or to prevent an uncontrolled release of radioactivity with radiological consequences in excess of the guidelines in Chapter 15.

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

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

3.2.4 Applicable General Design Criteria

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

- A. GDC 1, "Quality Standards and Records"
- B. GDC 2, "Design Basis for Protection Against Natural Phenomena"
- C. 10 CFR 50.55a, "Codes and Standards"
- D. 10 CFR 50, Appendix B, "Quality Assurance Requirements for Nuclear Power Plants and Reprocessing Plants"
- E. 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants"

3.2.5 Standards, Codes, and Guides

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

- A. Industry Standards and Codes

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

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components"

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"

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

3.2.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful 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 that require protection against wind and tornado loadings should consider the following guidelines.

3.3.1 Design Guidelines

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

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

3.3.2 Applicable General Design Criteria

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

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

3.3.3 Standards, Codes, and Guides

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

- A. Industry Standards and Codes

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

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

B. RDT Standards

None

C. Guides

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

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

3.3.4 Supplementary Information

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

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

B. Pending Industry Codes and Standards

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

3.3.5 Topics for Future Standards Development

None

3.4 WATER LEVEL (FLOOD) DESIGN

Safety class systems identified in Section 3.2 that require protection against flooding should consider the following guidelines.

3.4.1 Flood Design

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

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

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

3.4.2 Applicable General Design Criteria

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

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

3.4.3 Standards, Codes, and Guides

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

- A. Industry Standards and Codes

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

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

B. RDT Standards

None

C. Guides

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

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

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

3.4.4 Supplementary Information

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

A. Background Information

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

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

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

B. Pending Industry Standards and Codes

1. ANSI N515 (ANS 2.1), Draft, "Guidelines for Determining Tsunami Criteria for Power Reactor Sites"

3.4.5 Topics for Future Standards and Codes

None

3.5 MISSILE PROTECTION

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

The occurrence of missiles capable of damaging safety class facility components, systems, or structures and resulting in exposures in excess of the dose guidelines for the applicable facility operating condition should not exceed the probability guidelines for design basis accidents discussed in Chapter 15.

3.5.1 Missile Generated at the Reactor Facility

Among the facility-generated missiles to be considered as potential hazards at pool-type reactors are those arising from failure of experiments, experimental facilities, and high-speed rotating machinery. Examples of potential missiles are explosive irradiation samples, gas cylinders, fan blades, overhead cranes and handling equipment or tools that may be dropped into the reactor pool.

Equipment with a likelihood for generating missiles should be located such that the reactor core, stored fuel assemblies, and safety class systems will not be damaged by potential missiles. Where the physical separation of systems does not provide adequate missile protection, barriers or operating procedures should be used.

3.5.2 Missiles Generated by Natural Phenomena

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

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

3.5.3 Other Externally Generated Missiles

Aircraft crashes and explosions on nearby transportation routes may generate missiles against which safety class systems at a reactor facility should be protected. The peak overpressure resulting from an explosion may be equated to the dynamic wind pressure from the design basis tornado (see USNRC

Regulatory Guide 1.91). In most cases, 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 an overpressure that exceeds the equivalent tornado dynamic wind pressure.

3.5.4 Applicable General Design Criteria

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

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

3.5.5 Standards, Codes, and Guides

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

A. Industry Standards and Codes

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

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis"
2. USNRC Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites"

3.5.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," *ibid.*
3. USNRC Standard Review Plan 3.5.1.6, "Aircraft Hazards," *ibid.*
4. USNRC Standard Review Plan 3.5.3, "Barrier Design Procedures," *ibid.*
5. Branch Technical Position AAB 3-2, "Tornado Design Classification (Appended to SRP 3.5.1.4)," *ibid.*
6. WASH 1361, "Safety Related Site Parameters for Nuclear Power Plants," January 1975
7. Department of the Army Technical Manual TM5-1300, "Structures to Resist the Effects of Accidental Explosions," June 1969
8. IAEA Safety Guide SG-D5, Draft, "Man-Induced Events," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

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

3.5.7 Topics for Future Standards Development

None

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

Due to the very low energy of pool-type reactor fluid systems (low operating temperature and pressure), a gross piping rupture is unlikely. Dynamic effects of postulated pipe breaks would generally be minimal and protection would not be required.

For the purpose of evaluating the potential consequences of a Loss-of-Coolant-Accident (LOCA) in a pool-type reactor, a pipe break should be postulated as recommended in Section 5.1.

3.7 SEISMIC DESIGN

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

3.7.1 Seismic Input

The seismic input to the plant design should be based on a detailed seismic investigation of the reactor site which defines the site or facility specific seismic parameters, including design response spectra and critical damping values. These parameters are discussed below. The Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) should be defined to be consistent with the definitions in 10 CFR 100, Appendix A.

3.7.1.1 Design Response Spectra

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

3.7.1.2 Critical Damping Values

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

3.7.2 Seismic System Analysis

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

A. Equivalent static load analysis of systems (justified based on simple Category III reactor system configuration and conservative analytical technique).

B. Natural frequency and response load analysis, including the time-history of acceleration or response spectrum at the reactor operating location.

C. Soil-structure interaction analysis using the finite element technique, or equivalent, where foundations are deeply embedded in soil. Shallow foundations may use finite element or multiple mass spring (shear beam) methods. Structures supported on rock may use the fixed-base approach.

D. Floor response spectra should be developed from taking into consideration the three components of earthquake motion. The time-history method generally results in acceptable spectra. If a modal response spectra method is used, its conservatism should be verified.

E. Appropriate methods should be used to account for torsional effects and overturning moments.

F. The seismic analysis should account for damping in different elements of the model of a coupled system.

3.7.3 Seismic Instrumentation

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

3.7.4 Applicable General Design Criteria

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

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

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

3.7.5 Standards, Codes, and Guides

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

A. Industry Standards and Codes

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

B. RDT Standards

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

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

C. Guides

1. USNRC Regulatory Guide 1.12, "Instrumentation for Earthquakes"
2. USNRC Regulatory Guide 1.29, "Seismic Design Classification"
3. USNRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"
4. USNRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants"
5. USNRC Regulatory Guide 1.92, "Combining Modal Response and Spatial Components in Seismic Response Analysis"
6. USNRC Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components"

3.7.6 Supplementary Information

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

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

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

3.8.1 Concrete or Steel Confinement or Safety-Related Reactor Building

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

- A. Those loads encountered during preoperational testing.
- B. Those loads encountered during normal plant startup, operation, and shutdown.
- C. Those loads to be sustained during severe environmental conditions, including those induced by the design wind and the operating basis earthquake specified for the plant site.
- D. Those loads to be sustained during extreme environmental conditions, including those induced by the design basis tornado and the safe shutdown earthquake specified for the plant site.
- E. Those loads to be sustained during and after abnormal plant conditions, which includes a design basis Loss-of-Coolant Accident (LOCA).

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

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

Internal structures may have safety functions for which their structural integrity is important. For example, by providing support for overhead cranes during normal operations and during seismic disturbances, internal structures could prevent the occurrence of some internally generated missiles that may damage the reactor core or safety class components within the reactor building.

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

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

3.8.3 Other Seismic Category I Structures

Category III reactor facilities do not generally have other Seismic Category I structures. Most Category III reactors systems are within the reactor building.

3.8.4 Applicable General Design Criteria

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

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

B. GDC 4, "Environmental and Missile Design Bases"

3.8.5 Standards, Codes, and Guides

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

A. Industry Standards and Codes

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

2. ANSI A89.2-1972 (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 A169.1-72 (ACI 322-72), "Building Code Requirements for Structural Plain Concrete"

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

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

B. RDT Standards

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

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

C. Guides

1. USNRC Regulatory Guide 1.10, "Mechanical (Cadwell) Splices in Reinforcing Bars in Category I Structures"
2. USNRC Regulatory Guide 1.15, "Testing of Reinforcing Bars for Category I Concrete Structures"
3. USNRC Regulatory Guide 1.29, "Seismic Design Classification"
4. USNRC Regulatory Guide 1.55, "Concrete Placement in Category I Structures"
5. USNRC Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)"

3.8.6 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 3.8.5, "Foundations" *ibid.*
3. Teal, E.J., "Seismic Design Practice for Steel Buildings," AISC Engineering Journal, 4th Quarter, 1975

B. Pending Industry Standards and Codes

None

3.8.7 Topics for Future Standards Development

None

3.9 MECHANICAL SYSTEMS AND COMPONENTS

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

3.9.1 Safety Class Mechanical Components

Due to the basic design features of the open, unpressurized reactor coolant system, few Category III reactor safety class mechanical components are covered by the ASME Boiler and Pressure Vessel Code (this is the general reference for a Safety Class 1, 2, and 3 component in a commercial power reactor). If a particular Category III reactor facility should determine that some systems or components should be designed to the requirements of the ASME B&PV code, that facility should consult the Design Guide for Category II Reactors for the appropriate design guidelines.

Safety class mechanical components at Category III reactors may include the following:

- A. Reactor Pool Liner
- B. Fuel assemblies
- C. Portions of the reactor internal structures
- D. Mechanical portions of the reactivity control system
- E. ECCS pumps (if required)

The allowable stress, deformation and fatigue guidelines for non-ASME code mechanical components may be based on the criteria established in applicable industry codes for similar equipment, may be derived from manufacturer's standards, or may be derived from empirical methods based on field experience and/or testing. Such derived 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.

Table 3.9-1. Facility Operating Conditions.

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

These guidelines should ensure that mechanical components not subject to the ASME code are designed with sufficient conservatism to ensure the safe shutdown and cooling of the reactor and/or to prevent an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15.

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

Following the guidelines in USNRC Regulatory Guide 1.48 "Design Limits and Loading Combinations for Seismic Category I Fluid System Components," the loads resulting from the 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 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 reactor facility is sufficiently conservative to withstand upset or faulted events that may be 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 safety class mechanical components that must perform a mechanical motion in the performance of a safety function should be assured. These safety class mechanical components should be designed to perform their safety function under the environmental conditions discussed in Section 3.11. In addition, the operability of these mechanical components should be demonstrated by testing, detailed stress and deformation analyses or by a combination of these methods.

3.9.3 Non-Safety Class Mechanical Components

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

Table 3.9-2. Basis for ASME Boiler and Pressure Vessel Code Section III Service Loading Limits⁽¹⁾.

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

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

3.9.4 Applicable General Design Criteria

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

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

3.9.5 Standards, Codes, and Guides

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

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

3.9.6 Supplementary Information

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 draft are listed for information only.

- A. Background Information
 - 1. USNRC Standard Review Plan 3.9.6, "Inservice Testing of Pumps and Valves," NUREG 75/087, September 1975
- B. Pending Industry Standards and Codes
 - None

3.9.7 Topics for Future Standards Development

None

3.10 SEISMIC QUALIFICATION OF CATEGORY I INSTRUMENTATION AND ELECTRIC EQUIPMENT

Section 3.2 provides guidelines for determining if instrumentation and electrical equipment should be designed to Seismic Category I standards. If it is determined that instrumentation and electrical equipment should be Seismic Category I, the following qualification guidelines are applicable:

3.10.1 Seismic Qualification

Qualification of Category I instrumentation and electrical equipment generally includes tests and/or analysis to confirm operability during and after earthquakes of magnitude up to and including the SSE. When testing is the selected method of seismic qualification, the following guidelines are applicable:

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

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

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

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

3.10.2 Applicable General Design Criteria

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

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

3.10.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes

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

B. RDT Standards

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

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

C. Guides

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

3.10.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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 N178.2 (Series), Draft, "Requirements for Functional Qualification of Power-Driven Valve Actuators for Safety Related Applications in Nuclear Power Generating Stations"

3.10.5 Topics for Future Standards Development

None

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

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

Considering the spectrum of postulated accidents at Category III reactors (see Chapter 15), there is no mechanism (with the exception of LOCA-induced flooding described in Section 3.4) for large changes in the temperature, pressure or humidity conditions in the reactor building or in a new fuel storage vault. As a result, no special post-accident environmental design requirements are generally imposed on Category III reactor mechanical and electrical equipment.

If it should be determined that a particular Category III reactor facility requires qualification of mechanical and electrical equipment because of unique accident/post-accident environmental conditions, that facility should consult the Design Guide for Category II Reactors for environmental design guidelines.

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3.12 WELDING OF NUCLEAR COMPONENTS

3.12.1 Welding Guidelines

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

3.12.2 Applicable General Design Criteria

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

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

3.12.3 Standards, Codes, and Guides

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

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

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

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

C. Guides

None

3.12.4 Supplementary Information

None

3.12.5 Topics for Future Standards Development

None

3.13 NUCLEAR REACTOR FACILITY MATERIALS

3.13.1 Materials Guidelines

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

3.13.2 Applicable General Design Criteria

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

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

3.13.3 Standards, Codes, and Guides

Consult the Design Guide for Category II Reactors, "Light and Heavy Water Reactors," for a listing of materials - related standards, codes and guides that may also be applicable to Category III reactors.

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. USNRC Standard Review Plan 4.5.1, "Control Rod Structural Material," NUREG 75/087, September 1975
2. USNRC Standards Review Plan 4.5.2., "Reactor Internals Materials,"
ibid.
3. "Building Materials List," Underwriters Laboratory, Inc.,
4. MIL-F-51079B, Filter Medium, Fire Resistant, High Efficiency, 1974

B. Pending Industry Standards and Codes

None

3.13.5 Topics for Future Standards Development

None

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3.14 DECOMMISSIONING OF REACTOR FACILITIES

3.14.1 Decommissioning Guidelines

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

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

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

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

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

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

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

B. Reactor facility design should include provisions which will facilitate the eventual decommissioning of the facility and, in the process, will assist in maintaining occupational radiation exposures as low as reasonably achievable (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 reasonably achievable (see Chapter 12).

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

3.14.2 Applicable General Criteria

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

3.14.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes

None

B. RDT Standards

None

C. Guides

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

3.14.4 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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," International Atomic Energy Agency, Vienna, 1975

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

B. Pending Industry Standards and Codes

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

3.14.5 Topics for Future Standards Development

None

CHAPTER 4

REACTOR

4.1 REACTOR CORE ASSEMBLY

4.1.1 Scope

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

4.1.2 Safety Function

A. The reactor core should maintain fuel geometry and necessary coolant flow paths in order to ensure that the reactor can be safely shut down and adequately cooled during all facility operating conditions.

B. The fuel cladding should form the primary barrier against the release of fission products from the fuel.

4.1.3 Definitions

Reactor core assembly is defined as consisting of the following components located within the reactor coolant system pool:

- A. The Fuel and Fuel Assemblies
- B. In-core Reactivity Control Systems
- C. Core Support Structures
- D. In-core Experimental Facilities
- E. Reflector Elements

4.1.4 Design Guidelines

4.1.4.1 Guidelines for the Design of the Core Assembly

A. For normal and upset facility operating conditions (defined in Table 3.9-1), the reactor should be designed such that fuel cladding damage or fuel melting is not expected to occur during its design lifetime. The reactor should be designed to facilitate a rapid return to normal operation following an upset event. Fuel cladding damage is defined as perforation, excessive distortion, or rupture of fuel cladding as a direct result of a particular combination of power density and coolant conditions. It is assumed to occur if the

Figure 4.1-1. Guide to the Use of Section 4.1.

heat flux equals or exceeds the value corresponding to the onset of the transition from nucleate to film boiling (DNB) for a period long enough for cladding temperature to rise to the damage threshold appropriate to the specific fuel design.

B. For emergency and faulted facility operating conditions (defined in Table 3.9-1), the reactor should be designed such that any fuel or cladding damage that occurs shall not prevent reactor shutdown and adequate core cooling. Herein, "adequate core cooling" implies sufficient cooling to preclude additional damage to fuel and cladding following an emergency or faulted design basis accident; thus, fuel and cladding temperatures should be maintained conservatively below their respective melting points and metal-water reactions should be suitably limited.

C. The reactor core should be designed to inherently shutdown (due to loss of reflector and/or moderator) following a Loss-of-Coolant Accident (LOCA) that may uncover the reactor core. See Section 5.1 for the spectrum of LOCAs to be considered in Category III reactor design.

D. The potential effects of experimental facilities associated with a reactor core, including sample tubes or irradiation holes and their contents, 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.

E. The reactor core should incorporate thermal-hydraulic characteristics which ensure an adequate flow of coolant during all modes of operation.

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

G. Instrumentation should be provided to monitor reactor core power over the full range of power operation. Protection system (see Section 7.1) inputs should be provided, as required, to prevent core assembly damage in excess of the guidelines in A. and B. above.

H. The reactor core assembly should be protected from the effects of severe natural phenomena and missiles as described in Chapter 3. Since the reactor coolant system of pool-type reactors is generally open to the reactor building at the pool surface, the facility design may require additional structures, barriers, or procedures to ensure adequate protection of the core assembly from items that may be dropped or may fall into the reactor pool.

I. Mechanical components of the core assembly should consider the design guidance in Section 3.9.

4.1.4.2 Maintenance, Testing, and Inspection

Category III reactor cores should be designed to permit periodic visual inspection of the reactor core assembly using underwater inspection techniques.

4.1.4.3 Interfaces With Other Systems

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

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

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

4.1.5 Applicable General Design Criteria

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

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

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 also be considered in the design of the reactor core assembly:

- A. Industry Standards and Codes
 - 1. ANSI N18.3-1972, "Fuel Element Identification"
 - 2. ANSI N18.6 (ANS-5.1), Draft, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors"
 - 3. ANSI N147-1973, "Practice for Examination of Fuel Element Cladding, Including the Determination of Mechanical Properties"
 - 4. ANSI N398-1974, "Quality Control for Plate Type Uranium-Aluminum Fuel Elements"
 - 5. ANSI N411-1975, "Nuclear Data Sets for Reactor Design Calculations"

6. ANSI N412-1975, "The Determination of Neutron Reaction Rate Distributions and Reactivity of Nuclear Reactors"

7. ANSI N652-1976, "A Guide for Acquisition and Documentation of Reference Power Reactor Physics Measurements for Nuclear Analysis Verification"

8. ANSI N676-1976, "The Determination of Thermal Energy Deposition Rates in Nuclear Reactors"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 2.3, "Quality Verification for Plate-Type Uranium-Aluminum Fuel Elements for Use in Research Reactors"

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

1. USNRC Standard Review Plan 4.2, "Fuel System Design," NUREG 75/087, September 1975

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

3. USNRC Standard Review Plan 4.4, "Thermal and Hydraulic Design," *ibid.*

B. Pending Industry Standards and Codes

1. ANSI N41.22 (ANS 57.4), Proposed, "Failed Fuel Detection Systems"

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

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

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

5. ANSI N704 (ANS 19.5), Draft, "Requirements for Reference Integral Reactor Physics Measurements"

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

4.1.8 Topics for Future Standards Development

The following topical listing identifies areas in which the development of additional nuclear standards may provide a more consistent approach to Category III reactor facility design and modification.

A. Standard approach to reactor design and performance analysis for research and testing reactors to facilitate a consistent analytical approach to verify the adequacy of reactor core assembly, reactivity control system and in-core experimental facility design and modification.

4.2 REACTIVITY CONTROL SYSTEM

4.2.1 Scope

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

4.2.2 Safety Function

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

4.2.3 Definition

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

4.2.4 Design Guidelines

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

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

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

- a. Safe shutdown of the reactor.
- b. Limited fuel damage, per the guidelines in Section 4.1.
- c. Off-site radiological consequences of postulated accidents are limited to the guideline doses in 10 CFR 100.

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

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

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

Figure 4.2-1. Guide to the Use of Section 4.2.

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

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

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

4.2.4.2 Guidelines for the Design of a Reactivity Control System

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

B. Two independent and diverse reactivity control systems should be provided. These systems may utilize control rods, movable reflector elements or other suitable means of reactivity control

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

D. Generally, two reactivity control systems should be capable of reliably controlling reactivity changes during normal and upset operating conditions and those emergency events that may be expected during the life of the facility (defined in Chapter 15). However, inherent reactor safety features such as a large (and in some reactors, prompt) negative temperature coefficient and/or limited excess reactivity may effectively limit the consequences of postulated reactivity transients that may arise during the facility operating conditions discussed above, 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. The reactivity control system should be designed to "fail safe" on loss of power (electric, pneumatic, or hydraulic power, as appropriate). Therefore, the reactivity control system generally does not require Class IE electric power.

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

G. Potential accidents that can increase the reactivity of the core and redistribute power (see Chapter 15), including those arising from the operation

or failure of experimental facilities, should be evaluated to determine the response time, positive and negative reactivity addition rates, and other safety related aspects of the reactivity control system necessary to ensure that the fuel damage guidelines of Section 4.1 and the applicable reactor coolant system stress limits are not exceeded.

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

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

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

K. Safety class portions of the reactivity control system should be qualified for all anticipated environmental conditions where the specific environments are based on the spectrum of accidents in Chapter 15 (see Section 3.11).

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. The reactivity control system should be designed to permit periodic visual inspection.

C. Testing provisions of Sections 7.1 and 8.1 apply to safety-related electrical portions of the reactivity control system.

4.2.4.4 Interfaces With Other Systems

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

- A. Reactor Core Assembly
- B. Reactor Coolant System
- C. Protection System
- D. Non-Safety I&C System
- E. Non-Class IE Electric Power System

F. Fuel Handling and Storage 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 Application 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 12, "Suppression of Reactor Power Oscillation"

C. GDC 13, "Instrumentation and Control"

D. GDC 20, "Protection System Functions"

E. GDC 24, "Separation of Protection and Control Functions"

F. GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"

G. GDC 26, "Reactivity Control System Redundancy and Capability"

H. GDC 27, "Combined Reactivity Control Capability"

I. GDC 28, "Reactivity Limits"

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

4.2.6 Standards, Codes, and 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

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

2. USNRC Standard Review 4.4, "Thermal and Hydraulic Design," *ibid.*

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

B. Pending Industry Standards and Codes

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

4.2.8 Topics for Future Standards Development

The following topical listing identifies areas in which the development of additional nuclear standards may provide a more consistent approach to Category III reactor facility design and modification.

A. Standard approach to reactor design and performance analysis for research and testing reactors to facilitate a consistent analytical approach to verify the adequacy of reactor core assembly, reactivity control system and in-core experimental facility design and modifications.

CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED COOLANT SYSTEMS

5.1 REACTOR COOLANT SYSTEM

5.1.1 Scope

This section provides guidelines for the design of a reactor coolant system and for determining the minimum safety class of components of the system. Figure 5.1-1 is a guide to the use of this section.

5.1.2 Safety Function

The reactor coolant system should provide for adequate core cooling and thereby ensure that the fuel design guidelines of Section 4.1 are not exceeded during any facility operation condition.

5.1.3 Definition

The reactor coolant system is defined as those components which form the primary coolant system boundary, which includes the following:

- A. Reactor pool structure and liner.
- B. The primary side (generally the tube side) of the main heat exchangers, (if provided).
- C. The pumps, valves, and piping completing the fluid flow paths for removing heat from the core and rejecting it to the cooling water system (if provided).
- D. Other attached piping out to a suitable boundary (e.g., purification system and make-up water system piping and instrument lines out to a suitable isolation valve).

5.1.4 Design Guidelines

5.1.4.1 Guidelines for Determining the Minimum Safety Class of the Reactor Coolant System

Considering the guidelines of 10 CFR 50.55a (that structures, systems, and components be designed, fabricated, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed) it can be seen that the determination of the safety class of reactor coolant system may be flexible to an extent determined by the potential consequences of postulated accidents. Unlike the reactor coolant system of power reactors, the reactor coolant system boundary of Category III reactors is not a

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Figure 5.1-1. Guide to the Use of Section 5.1.

Figure 5.1-1. Guide to the Use of Section 5.1 (Continued).

pressure retaining boundary that forms a physical barrier to the release of fission products from the reactor core. It is therefore, unlikely that a Category III reactor would require components of the reactor coolant system to be designed to the same quality standards applicable to power reactors (e.g., Safety Class 1). It is recommended that Category III reactor facilities be designed to adequately protect health and safety with a Safety Class 3 or non-nuclear reactor coolant system boundary, based on the guidelines discussed below.

Loss of coolant experiments have been conducted on the ASTR and LITR reactors (References 1-3). In general, these experiments have confirmed that the maximum fuel plate temperature following a loss of all reactor coolant will not cause melting of aluminum clad fuel plates of a reactor operating continuously at 1 MWt with thermal flux and power density similar to LITR (average thermal flux of 2×10^{13} n/cm²-sec, maximum thermal flux of 3.5×10^{13} n/cm² sec and average power density of 31.4 Kw/liter - see Reference 4). Based on the data presented in References 1-3, 1 MWt reactors operating with average thermal flux and power densities similar to LITR should withstand the effects of a loss-of-coolant accident without significant fuel or cladding damage. References 1-3 additionally indicate that fuel melting did not occur during the loss-of-coolant tests on the ASTR and LITR reactors when operated above 1 MWt with limited power histories. Where power histories of reactors can be suitably limited by technical specification limits, fuel and cladding damage during a LOCA can be prevented. Unlike power reactors, which may release large quantities of fission products to the containment following a LOCA (see References 5 and 6), Category III reactor facilities, that can withstand the effects of a LOCA with minimal damage, will release a much smaller amount of radioactive material to the reactor building following a LOCA because the fuel cladding remains an effective barrier against the release of fission products from the fuel and/or because the fission products are retained in the fuel matrix. The resulting smaller source terms to be considered in evaluating the potential radiological consequences of a LOCA include activity in the primary coolant and fission product activity released from a small portion of the fuel that has experienced minor cladding failures during normal reactor operation. As indicated in Table 5.1-1, the potential consequences of the LOCA form the basis for determining the appropriate design and quality standards applicable to reactor coolant system boundary components. If the reactor facility can withstand a LOCA but the potential off-site doses exceed 0.5 rem, it is recommended that the minimum safety class of the appropriate reactor coolant system components be Safety Class 3. As used herein, the term "withstand" implies the following:

A. No fuel damage in excess of the guidelines of Section 4.1 (and, if spent fuel is stored in the reactor pool, no spent fuel damage in excess of the guidelines of Section 9.2).

B. Stress in a safety class mechanical component is limited to the applicable stress allowable guidelines (see Section 3.9).

C. No safety class system or component is rendered unable to complete its required system safety function as a result of the LOCA (e.g., due to flooding).

Table 5.1-1. Minimum Reactor Coolant System
Component Safety Class Recommendations

Are the Nuclear, Mechanical, and Thermal-Hydraulic Consequences of a LOCA Resulting from a Failure of the Component Acceptable?	Are the Maximum Radiological Consequences of a LOCA Resulting from a Failure of the Component <0.5 rem Off-Site?	Minimum Recommended Safety Class of the Reactor Coolant System Component
Yes	Yes	Non-Nuclear
	No	SC-3
No	--	

The reactor coolant system, or portions of the reactor coolant system, may be designed to non-nuclear standards if the LOCA caused by their failure can be withstood by the facility and if the potential off-site radiological consequences do not exceed 0.5 rem.

5.1.4.2 Guidelines for the Design of a Reactor Coolant System

A. The reactor coolant system boundary should be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage or of rapidly propagating failure.

B. Drains, systems or piping permanently connected to the reactor pool or other features that by maloperation or failure (e.g., pipe break) would cause a loss of coolant from the reactor pool that would uncover the reactor core or spent fuel stored in the reactor pool should not be included in the design of the reactor coolant system if the consequences of such a LOCA would exceed the guidelines in Chapter 15.

C. If a safety class reactor coolant system is required, it should be designed as a Seismic Category I system.

D. Mechanical components of the reactor coolant system should consider the design guidance in Section 3.9.

E. The reactor coolant system should be protected against the effects of missiles (see Section 3.5).

F. Reactor heat removal during normal operation may utilize natural or forced circulation of the reactor coolant, as described below:

1. Natural circulation with heat rejection to passive heat sinks is acceptable if reactor coolant bulk temperature is maintained at acceptably low values (maximum bulk temperature limited to 120 F - 140 F) and the release of short-lived gaseous nuclides (e.g., ^{16}N , ^{41}Ar) to the reactor building atmosphere do not result in occupational doses in excess of the guidelines in Chapter 12.

2. Natural circulation with heat rejection to an in-pool heat exchanger (to enhance natural circulation flow) may be acceptable for reactor heat loads beyond the capability of passive heat sinks if the temperature and dose guidelines in F.1 above are satisfied.

3. Forced circulation with heat rejection to an ex-pool heat exchanger and reactor coolant system loop piping may be necessary to permit higher power reactor operation or to reduce the occupational dose resulting from the short-lived gaseous nuclides (e.g., ^{16}N , ^{41}Ar) by means of holdup tanks or other suitable features that provides time for the decay of the nuclides before returning the reactor coolant to the pool.

4. Other configurations may also be acceptable if the temperature and dose guidelines in F.1 are satisfied.

G. The Loss-of-Coolant Accidents (LOCAs) for which the Category III reactor facility should be designed include the severence of a spectrum of piping connected to the reactor pool. As used herein, "severence" of a pipe implies a guillotine-type break of a pipe (e.g., due to an earthquake) and the subsequent loss of water from the reactor pool to the lowest level possible for the particular pool and fluid system configuration. The piping breaks to be considered include the following:

1. The largest diameter piping (e.g., reactor coolant system loop piping, if provided).
2. The piping that results in the greatest loss of water from the pool (e.g., a drain line).
3. Other piping, the failure of which may cause a LOCA.

Credit may be taken for siphon breaks that can be shown to be effective in preventing siphon draining of the reactor pool and for other fluid system geometries that may prevent lowering the level of the pool water below the level of the reactor core.

Generally a LOCA at a Category III reactor facility will be comparable to draining the pool by gravity through an appropriately sized line. For those facilities having reactor coolant system loop piping external to the pool, a LOCA on the discharge side of the reactor coolant pump should be postulated and the effects of the pumped loss of water from the pool should be determined.

It is not necessary to postulate a LOCA due to a break in a safety class reactor pool liner. A reactor pool liner designed to non-nuclear standards should, however, be considered as a potential LOCA source. The size and location of the postulated break should be based on a mechanistic evaluation of reactor coolant system response to a Safe Shutdown Earthquake (SSE).

The above spectrum of potential loss-of-coolant accidents should be considered in establishing design and performance requirements for an ECCS, if required, (see Section 6.1) and for the make-up water system (see Section 9.3).

H. Reactor coolant system components do not require Class IE power. They need to be operable on off-site power only.

I. Reactor coolant chemistry controls should be instituted as necessary to:

1. Minimize reactor coolant system general and localized corrosion.
2. Minimize the amount of corrosion products available for transport by the primary coolant and for activation and subsequent deposition throughout the reactor coolant system.
3. Minimize fouling of heat transfer surfaces (fuel elements and heat exchanger tubes).

4. Suppress radiolytic dissociation of water.

5. Limit reactor coolant circulating activity levels.

J. Instrumentation should be provided in the reactor coolant system to ensure that system parameters are maintained within safety limits (e.g., pool water level).

5.1.4.3 Maintenance, Testing, and Inspection

A. The reactor coolant system should be designed to facilitate periodic in-service inspection of the reactor pool and of equipment within the pool.

B. Reactor pool structural material test specimens for a material surveillance program should be accommodated in the zone of maximum radiation exposure of the pool structure. The irradiated test specimens should be capable of removal and testing to establish the material properties of the irradiated reactor pool structural components. No material surveillance program is required for ferritic vessels which will be exposed to peak neutron fluences ($E > 1$ MeV) less than 10^{17} n/cm² at the end of design life. Requirements for materials surveillance programs for non-ferritic materials should be based on similarly conservative guidelines.

5.1.4.4 Interfaces with Other Systems

The reactor coolant system may interface with other facility systems, including the following:

- A. Reactor Core Assembly
- B. Reactivity Control System
- C. Emergency Core Cooling System
- D. Protection System
- E. Non-Safety I&C Systems
- F. Non-Class IE Electric Power System
- G. Cooling Water System
- H. Fuel Handling and Storage System
- I. Purification and Make-up Water Systems
- J. Experimental Facilities
- K. Radioactive Waste System

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

5.1.5 Applicable General Design Criteria

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

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

5.1.6 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 a reactor coolant system:

A. Industry Standards and Codes

- 1. ANSI N18.6 (ANS 5.1), Draft, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors"
- 2. ANSI N146-1973 (ASTM E185-73), "Surveillance Tests on Structural Materials in Nuclear Reactors"
- 3. ANSI B96.1-1973, "Specification for Welded Aluminum-Alloy Field Erected Storage Tanks"
- 4. API 620, "Recommended Rules for Design and Construction of Large, Welded Low Pressure Storage Tanks," American Petroleum Institute, 1973
- 5. API 650, "Welded Steel Tanks for Oil Storage," American Petroleum Institute, 1973
- 6. AWWA D100-73, "Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage," American Water Works Association, 1973

B. RDT Standards

None

C. Guides

None

5.1.7 Supplementary Information

None

5.1.8 Topics for Future Standards Development

The following topical listing identifies areas in which the development of additional nuclear standards may provide a more consistent approach to Category III reactor facility design and modification:

A. Guidelines for material surveillance programs for non-ferritic reactor coolant system material.

5.1.9 Section 5.1 References

1. Cox, J.A. and Webster, C.C., "Water-Loss Test at the Low Intensity Testing Reactor," ORNL-TM-632, Oak Ridge National Laboratory, August 1964
2. Webster, C.C., "Water Loss Tests in Water Cooled and Moderated Research Reactors," Nuclear Safety, Vol. 8, No. 6, p. 590, Nov-Dec 1967
3. The Technology of Nuclear Reactor Safety, Volume 1, Section 5.3.2, "Fuel Temperatures Reached on Loss of Coolant in the LITR"
4. Directory of Nuclear Reactors, Volume II, p. 81, "Low Intensity Test Reactor," International Atomic Energy Agency, Vienna, 1959
5. Di Nunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962
6. USNRC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for PWRs"

5.2 SECONDARY COOLANT SYSTEM

A Category III reactor does not require a secondary coolant system. If active cooling of the reactor coolant is required to permit sustained high power reactor operation, that function is performed by the cooling water system as described in Section 9.1.

5.3 RESIDUAL HEAT REMOVAL SYSTEM

The safety function of an RHR system is to transfer fission product decay heat and other residual heat from the reactor core after reactor shutdown at a rate such that the specified acceptable fuel design limits (see Section 4.1) and design conditions of the reactor coolant system boundary are not exceeded during periods of long term shutdown cooling.

Category III reactors can generally accomplish this function by rejecting the residual and decay heat from a shutdown reactor to passive heat sinks (e.g., the pool water, pool structure and the reactor building atmosphere) without the operation of active systems or components. Category III reactors, therefore, do not generally require a residual heat removal system.

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CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 EMERGENCY CORE COOLING SYSTEM

6.1.1 Scope

This section provides guidelines for determining the need for an Emergency Core Cooling System (ECCS). In addition, basic guidelines relevant to the design of an emergency core cooling system are provided. Figure 6.1-1 is a general guide to the use of this Section

6.1.2 Safety Function

The ECCS should transfer heat from the reactor core following a loss of reactor coolant which exceeds the makeup capacity of the makeup water system (see Section 9.3) at a rate such that fuel and clad damage are limited per the guidelines in Section 4.1.

Where this function can be performed by entirely passive means (e.g., conductive heat transfer via reactor internal structures), no ECCS is required.

6.1.3 Definition

The emergency core cooling system is defined as those components (including pumps, valves, piping, and spray nozzles) which must function to provide water to cool the reactor core following loss of coolant accidents. The ECCS includes the sources of water necessary to accomplish the safety function. Instrumentation, control, and electrical systems which initiate or support ECCS operation should consider the guidelines of Sections 7.1 and 8.1.

6.1.4 Design Guidelines

6.1.4.1 Guidelines for Determining the Need for an ECCS

This Design Guide divides Category III reactors into two groups as described in Chapter 1. The ECCS requirements for each group is discussed below and summarized in Figure 6.1-1.

If it is determined that an ECCS is required in order to assure acceptable consequences of a LOCA (defined in Subsection 5.1.4), it should be designed to a minimum safety class of SC-3 to be compatible with the safety class requirements of the reactor coolant system (see Section 5.1). Guidelines for the design of an ECCS are given in Subsection 6.1.4.2.

A. Reactors with a Maximum Power Level ≤ 1 Mwt.

Figure 6.1-1. Guide to the Use of Section 6.1.

References 1-3 summarize loss-of-coolant tests that were conducted on the ASTR and LITR reactors. In general, these experiments have confirmed that the maximum fuel plate temperature following a loss of all reactor coolant will not cause melting of the aluminum fuel plates of a reactor operating continuously at 1 MWt with thermal flux and power density similar to LITR (average thermal flux of 2×10^{13} n/cm²-sec, maximum thermal flux of 3.5×10^{13} n/cm²-sec and average power density of 31.4 kw/liter - see Ref. 4). Based on the data presented in References 1-3, reactors not exceeding the LITR operating parameters generally do not require an ECCS. Reactors substantially exceeding the LITR flux and power density levels should be individually examined to determine the maximum fuel plate temperatures following a Loss-of-Coolant-Accident (LOCA). If the temperature transient indicates that fuel or cladding damage is likely, the DOE facility contractor should assure that the LOCA fuel temperature transient is suitably limited by any of the following means:

1. A technical specification limit on reactor power history and, thus, on the decay heat rate.
2. Installing an ECCS with suitable heat removal capability. For these small reactors, a core spray system may provide adequate core cooling based on the tests summarized in Reference 2.
3. Other suitable means for ensuring adequate core cooling.

B. Reactors With a Maximum Power Level:

$$1 \text{ MWt} < P \leq 10 \text{ MWt}$$

References 1-3 additionally indicate that fuel melting did not occur during the loss-of-coolant tests on the ASTR and LITR reactors when operated above 1 MWt with limited power histories. Where power histories can be suitably limited by technical specification limits, or where unique design or operating characteristics provide for effective core cooling, fuel and cladding damage in excess of the guidelines of Section 4.1 can be prevented during a LOCA and thus no ECCS may be required. Unique design or operating characteristics and the substantiating tests and/or analysis should be reviewed on a case basis to ensure the availability of adequate post-LOCA core cooling.

An ECCS should be provided for other Category III reactors that can not otherwise ensure adequate post-LOCA core cooling.

6.1.4.2 Guidelines for the Design of an ECCS

A. The ECCS should be capable of adequately cooling the reactor core following the spectrum of postulated loss-of-coolant accidents (see Section 5.1) that exceed the make-up capacity of the reactor coolant make-up water system (see Section 9.3).

B. The required capacity of the ECCS should include consideration of the core residual and decay heat following extended operation at a conservatively determined maximum expected steady-state power level in order to account for instrument error.

The decay heat generation rate should be based on the ANSI N18.6 standard fission product decay heat curve for uranium fueled thermal reactors (see Figure 6.1-2) or other similarly conservative method. The ANSI N18.6 decay heat curve is based on the hypothetical case of a ^{235}U , ^{238}U fueled thermal reactor that has operated for an infinite period of time at a constant power and is then instantaneously shut down. The curve may be corrected for shorter operating time.

ANSI N18.6 estimates the uncertainty of the results obtained using the standard curve as follows:

Time Since Instantaneous Shutdown	Uncertainty
$t < 10^3 \text{ sec}$	+20%, -40%
$10^3 \text{ sec} < t < 10^7 \text{ sec}$	+10%, -20%
$t > 10^7 \text{ sec}$	+25%, -50%

C. Category III reactors may generally utilize manual means for ECCS actuation due to their following inherent design features.

1. The atmospheric pressure reactor coolant system provides a low driving head for a LOCA. A LOCA in a Category III reactor is comparable to draining the reactor pool by gravity.

2. A large volume of water is generally available in the reactor coolant system to provide a significant time between the detection of a LOCA and the uncovering of the reactor core.

3. Residual heat rate is low due to the low operating temperature of Category III reactors (reactor coolant system maximum bulk temperature typically is less than $120^\circ - 140^\circ\text{F}$).

4. Relatively low decay heat generation rate due to low operating power level.

5. With a timely reactor trip following the detection of a LOCA, significant core decay heat is transferred to the water remaining in the reactor pool before the core becomes uncovered (see Refs. 1 and 2) thereby, limiting decay heat available for post-LOCA fuel heating.

6. Without a reactor trip, a LOCA will influence the reactivity of the reactor core due to loss of reflector and/or moderator and will generally shut-down the reactor (see Refs. 1 and 3).

D. Instrumentation and controls associated with the ECCS, including provisions for manual initiation, should meet the requirements specified in Section 7.1 for protection systems.

Figure 6.1-2. Standard Fission-Product Decay Heat Curve for Uranium-Fueled Thermal Reactors (Ref. ANSI N18.6)

E. The ECCS should be designed as a Seismic Category I system.

F. Mechanical components of the ECCS should consider the design guidance in Section 3.9.

G. Active ECCS components should be capable of operating on both off-site Class IE power (assuming one is unavailable).

H. ECCS components should be protected from the effects of missiles (see Section 3.5).

I. The ECCS design should incorporate suitable redundancy to permit the required safety functions to be completed following a LOCA with the addition of a single failure of a component in the ECCS. Application of the single failure criterion to system design is discussed in Chapter 15.

J. The ECCS design should incorporate suitable independence and separation such that a failure in one portion of the ECCS does not prevent the remaining portions from completing their required system safety functions when needed.

K. Analysis should be provided to demonstrate that peak cladding temperatures, maximum cladding oxidation, and maximum hydrogen generation rates are acceptable following a LOCA and subsequent successful ECCS operation. Additionally, the analysis should demonstrate that the core geometry remains coolable during the accident and for the extended period required for the decay of the long-lived radioactivity remaining in the core. Passive heat sinks can generally provide for long term post-LOCA core cooling. As described in Section 5.3, an active Residual Heat Removal System is generally not required.

6.1.4.3 Maintenance, Testing, and Inspection

A. The ECCS should be designed to permit operational testing of the system during periods when the reactor is shut down.

B. Maintenance during power operations which may temporarily disable a portion of the ECCS may be permitted subject to suitable technical specification restrictions on reactor operation during that period.

6.1.4.4 Interfaces With Other Systems

The ECCS may interface with the following other facility systems:

A. Reactor Coolant System

B. Protection System

C. Class IE Electric Power System

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

6.1.5 Applicable General Design Criteria

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

- A. GDC 35, "Emergency Core Cooling"
- B. GDC 36, "Inspection of Emergency Core Cooling System"
- C. GDC 37, "Testing of Emergency Core Cooling System"

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 an ECCS:

A. Industry Standards and Codes

- 1. ANSI N18.6 (ANS 5.1), Draft, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors"

B. RDT Standards

None

C. Guides

- 1. USNRC Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems"

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. Branch Technical Position EICSB 4, "Requirements for Motor Operated Valves in the ECCS," (in SRP appendix 7A), NUREG 75/087, September 1975

- 2. Cottrell, W.B., "A Selected Bibliography on Emergency Core Cooling Systems for Light-Water Cooled Reactors (LWR's), "ORNL-NSIC-113, January 1974

B. Pending Industry Standards and Codes

None

6.1.8 Topics for Future Standards Development

The following topical listing identifies areas in which the development of additional nuclear standards may provide a more consistent approach to Category III reactor facility design and modification:

A. Guidelines for the design of Emergency Core Cooling Systems for small nuclear reactors.

B. Standard Performance Criteria for an Emergency Core Cooling System for use in water cooled reactors having a fuel system other than zirconium clad UO_2 pellets (guidelines similar to 10 CFR 50.46 and 10 CFR 50, Appendix K).

6.1.9 Section 6.1 References

A. Cox, J.A. and C.C. Webster, "Water-Loss Test at the Low Intensity Testing Reactor," ORNL-TM-632, Oak Ridge National Laboratory, August 1964

B. Webster, C.C., "Water Loss Tests in Water Cooled and Moderated Research Reactors," Nuclear Safety, Vol. 8, No. 6, p. 590, Nov-Dec 1967

C. The Technology of Nuclear Reactor Safety, Volume 1, Section 5.3.2, "Fuel Temperatures Reached on Loss of Coolant in the LITR"

D. Directory of Nuclear Reactors, Volume II, p. 81, "Low Intensity Test Reactor," International Atomic Energy Agency, Vienna, 1959

6.2 REACTOR BUILDING

6.2.1 Scope

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

6.2.2 Safety Function

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

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

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

6.2.3 Definitions

The various types of reactor buildings are defined below:

A. Containment

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

B. Confinement

A confinement is defined as those structures and components (including the building structure, ventilation paths, filtration and adsorption units and fans) which enclose the reactor and associated systems and must function in order to control the release of radioactivity to the environment by means of a slight negative pressure within the building and a directed building exhaust

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Figure 6.2-1. Guide to the Use of Section 6.2.
6-7(a)

Figure 6.2-1. Guide to the Use of Section 6.2 (Continued).

Table 6.2-1. Safety Functions of Various Types of Reactor Buildings

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

6-7(c)

path which provides means for reducing the quantity of radionuclides released to the environment and for enhancing atmospheric dispersion. The confinement also protects enclosed systems against the effects of severe natural phenomena and missiles.

C. Safety-Related Reactor Building

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

D. Non-Safety Related Reactor Building

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

6.2.4 Design Guidelines

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

The following guidelines describe the basis for selecting a containment, confinement, safety-related reactor building or non-safety reactor building to house a reactor and associated systems. The need for a containment or confinement will be based on the potential radiological impact of the facility. The need for providing protection to enclosed systems or components against the effects of severe natural phenomena and missiles will be based on the protection requirements of the enclosed systems. The minimum safety class guidelines for the reactor building are summarized in Table 6.2-2. Design guidelines are presented in Subsections 6.2.4.2 through 6.2.4.7.

A. Containment and Confinement

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

Table 6.2-2. Minimum Safety Class Recommendations for Reactor Buildings

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

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

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

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

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

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

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

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

6.2.4.2 Guidelines for the Design of a Containment and Associated Systems

See the Design Guide for Category II Reactors.

6.2.4.3 Guidelines for the Design of a Confinement and Associated Systems

A. Confinement Structure and Exhaust System

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

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

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

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

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

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

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

B. Confinement Isolation System

No Confinement Isolation System is generally required. The spectrum of postulated accidents (see Chapter 15) do not generally cause a significant overpressure in the reactor building; therefore, a negligible leakage will occur through the confinement penetrations.

C. Confinement Heat Removal System

Considering the spectrum of postulated accidents (see Chapter 15), there is no mechanism for rapidly increasing pressure and temperature in the reactor building to a level that would require a heat removal capability beyond the passive heat sinks normally available within the reactor building. No confinement heat removal system is generally required.

D. Confinement Cleanup Systems

A confinement cleanup system (a recirculating system within the confinement) is generally not required since the dose guidelines specified in Chapter 15 can generally be met without this capability.

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

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

B. The safety-related reactor building should provide protection to enclosed systems from the effects of severe natural phenomena and missiles (see Chapter 13).

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

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

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

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

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

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

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

6.2.4.7 Interfaces with other Systems

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

- A. Emergency Core Cooling System
- B. Non-Safety I&C System
- C. Non Class IE Electric Power System
- D. Cooling Water 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.

6.2.5 Applicable General Design Criteria

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

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

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

None

6.2.6 Standards, Codes, and Guides

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

A. Industry Standards and Codes

1. Confinements

- a. ANSI A158.1-1970 (ACI 307-69), "Specification for the Design and Construction of Reinforced Concrete Chimneys"
- b. ANSI N18.6 (ANS 5.1), Draft, "Decay Energy Release Rate Following Shutdown of Uranium Fueled Thermal Reactors"
- c. ANSI/ASME N509-1976, "Nuclear Power Plant Air Cleaning Units and Components"
- d. ANSI N5101975, "Testing of Nuclear Air Cleaning Systems"
- e. ANSI/UL 586-1977, "Safety Standard for Test Performance of High Efficiency Particulate Air Filter Units"

2. Safety-Related Reactor Building

None

3. Non-Safety Related Reactor Buildings

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

- a. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures"
- b. ANSI A89.1-1972 (ACI 318-71), "Building Code Requirements for Reinforced Concrete"
- c. ANSI A138.1-1972 (ACI 301-72), "Specifications for Structural Concrete for Buildings"
- d. ANSI A169.1-1972 (ACI 322-72), "Building Code Requirements for Structural Plain Concrete"

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

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

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

B. RDT Standards

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

1. Confinements

a. E9-1T, "HEPA Filters"

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

None

C. Guides

1. Confinements

a. USNRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"

b. 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"

c. USNRC Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray"

d. 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"

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

2. Safety-Related Reactor Buildings

It is recommended that the following previously listed Regulatory Guides be considered in the design of a safety related reactor building:

a. Regulatory Guide 1.7

b. Regulatory Guide 1.82

3. Non-Safety Related Reactor Buildings

None

6.2.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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," NUREG 75/087, September 1975
2. UL-900; "Air Filter Units," Underwriter's Laboratories, Inc.
3. Burchsted, C.A., et al., "Nuclear Air Cleaning Handbook," ERDA 76-21, Oak Ridge National Laboratory, March 31, 1976.
4. MIL-F51068D, "Filter, Particulate, High Efficiency, Fire Resistant"
5. "Leakage Characteristics of Openings for Reactor Housing Components," NAA-SR-MEMO-5173, Atomics International, June 20, 1960
6. Kootz, R.L., et al., "Conventional Buildings for Reactor Containment," NAA-SR-10100, Atomics International, July 25, 1965
7. Kootz, R.L., et al., "Leakage Characteristics of Conventional Buildings Components for Reactor Housing Construction," Trans. Am. Nuc. Soc., November 1961

B. Pending Industry Standards and Codes

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

6.2.8 Topics for Future Standards Development

None

6.3 ENGINEERED SAFETY FEATURE VENTILATION SYSTEMS

6.3.1 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 6.3-1 is a guide to the use of this section.

6.3.2 Safety Function

An ESF ventilation system should protect on-site personnel in the control room and in other normally occupied enclosed areas of the reactor 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).

6.3.3 Definition

The ESF ventilation system is defined as the heating ventilation, and air conditioning systems or portions thereof (including associated duct work, dampers, fans and filter and adsorber systems) which must perform their safety functions 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.

6.3.4 Design Guidelines

6.3.4.1 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).

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. The accident/post-accident environmental conditions of the Category III reactor facility will generally remain within the appropriate limits for human habitability and/or for safety class equipment qualification (see Section 3.11).

Figure 6.3-1. Guide to the Use of Section 6.3.

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Figure 6.3-1. Guide to the Use of Section 6.3 (Continued).

An ESF ventilation system is, therefore, not generally required for post-accident environmental control.

C. The minimum safety class for the ESF ventilation system is Safety Class 3.

6.3.4.2 Guidelines for the Design of an ESF Ventilation System

A. 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.

B. The ESF ventilation system should be Seismic Category I.

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

D. This system should be capable of performing the required system safety functions of off-site or on-site electric power (assuming one is unavailable). The system should be supplied with Class IE electric power.

E. The ESF ventilation system should be protected against the effects of severe natural phenomena and missiles as described in Chapter 3.

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

G. The system should incorporate sufficient redundancy such that the system safety function can be accomplished with a single failure of an active component. Application of the single failure criterion to system design is discussed in Chapter 15.

H. The design of an ESF ventilation system should consider the guidelines in USNRC Regulatory Guide 1.52. Demisters preceding the filter banks may not be required if the relative humidity in the reactor building following postulated accidents is not high enough to impair the operation of the remaining filter

elements, which include a prefilter (prefiltering may be performed by the demister) followed by HEPA filters before an iodine adsorber bed (charcoal, metal zeolite or equivalent adsorber). HEPA filters should be provided after the iodine adsorber bed, if necessary, to remove additional particulates (e.g., charcoal or zeolite fines) from the exhaust stream.

6.3.4.3 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 visual inspection of the system and associated components.

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

6.3.4.4 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.

6.3.5 Applicable General Design Criteria

None

6.3.6 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/ASME 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 Particulate Air Filter Units"

B. RDT Standards

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

1. E9-1T, "HEPA Filters"

C. Guides

1. USNRC Regulatory Guide 1.52, "Design Testing and Maintenance Criteria for Engineered Safety Feature 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 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"
4. USNRC Regulatory Guide 3.2, "Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles"

6.3.7 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-586, "High Efficiency, Particulate, Air Filter Units," Underwriter's Laboratories, Inc.

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

7. Burchsted, C.A., et al., "Nuclear Air Cleaning Handbook," ERDA 76-21, Oak Ridge National Laboratory, March 31, 1976

8. Griwatz, G.H., Friel, J.V., and Bicehouse, J.L., "Entrained Moisture Separators for Fine (1-10 μ) Water-Air-Steam Service: Their Performance, Development, and Status," MSAR 71-45, Mine Safety Appliances Research Corporation.

9. WASH-1287, "Manual of Respiratory Protection Against Airborne Radioactive Material," USAEC 1974

10. MIL-F-51068D, "Filter, Particulate, High Efficiency, Fire Resistant," 1974

B. Pending Industry Standards and Codes

1. ANSI N189 (ANS 59.2), Proposed, "Safety Related Ventilation Systems"

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

6.3.8 Topics for Future Standards Development

The following topical listing identifies areas in which the development of additional nuclear standards may provide a more consistent approach to Category III reactor 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., toxic gas and airborne radiation hazards to operators and/or experiments inside the reactor building).

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CHAPTER 7

INSTRUMENTATION AND CONTROL

7.1 PROTECTION SYSTEM AND SAFETY-RELATED INSTRUMENTATION

7.1.1 Scope

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

7.1.2 Safety Function

A. The protection system should initiate protective actions based on the detection of abnormal plant conditions in order to safely shut down and cool 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-related instrumentation should provide the facility operator with sufficient information to readily determine the status of the protection system, operate the reactor within the prescribed range and initiate safety-related actions when necessary.

7.1.3 Definition

A. The protection system includes electrical and mechanical devices (from measured process variables to the input terminals of the system that performs 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 related instrumentation includes displays of safety related reactor facility parameters (generally derived from the same sensors used by the protection system) and protection system status (e.g., tripped, not tripped, interlocked, bypassed, operable) that are required in order to provide the operator with sufficient information to operate the reactor within the prescribed range and to perform required manual safety functions should such actions be necessary.

Figure 7.1-1. Guide to the Use of Section 7.1.

7.1.4 Design Guidelines

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

A protection system should be provided if automatically or manually initiated protective actions must be taken in order to assure safe reactor shutdown and cooling 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. Stress to Safety Class Mechanical Components in Excess of the Applicable Guidelines (Section 3.9).
- D. Radiation and Toxic Gas Exposure Guidelines for On-site Personnel (Section 6.3, Chapter 15).
- E. Off-Site Radiological Consequences Guidelines (Chapter 15).

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

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

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

7.1.4.2 Guidelines for the Design of the Protection System

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

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

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

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

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

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

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

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

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

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

J. Protection system instrumentation and controls sufficient for placing the reactor in a safe shutdown condition should be provided in a location remote from the control room for use during any condition which may make the control room uninhabitable (e.g., fire). As a minimum, this should include the ability to scram the reactor from a remote location.

K. Protection system instrument trip setpoints should be established with sufficient margin between the technical specification safety limit for the process variable and the nominal trip setpoint so that automatic protective

actions will correct an abnormal situation before a safety limit is exceeded. This margin should allow for the following:

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

L. The protection system should be Seismic Category I and should be qualified per Section 3.10.

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

N. Protection systems should generally be supplied with Class IE electric power. Class IE power for the protection system may not be required in the special case where loss of all electric power places the reactor in a safe shutdown condition and no active, electric-powered systems are required for shutdown cooling of the reactor or for preventing an uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15. This provision may apply to the smaller Category III reactors that do not require an ECCS or a confinement (which would require an active ventilation system to maintain reactor building negative pressure).

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

7.1.4.3 Maintenance, Testing, and Inspection

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

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

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

7.1.4.4 Interfaces With Other Systems

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

- A. Reactor Core Assembly

- B. Reactivity Control System
- C. Reactor Coolant System
- D. Emergency Core Cooling System
- E. Reactor Building
- F. ESF Ventilation System
- G. Non-Safety I&C System
- H. Class IE Electric Power System
- I. Fuel Handling and Storage System
- J. Fire Protection System
- K. Experimental Facilities
- L. 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 General Design Criteria

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

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

7.1.6 Standards, Codes, and Guides

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

A. Industry Standards and Codes

1. IEEE 279-1971 (ANSI N42.7-1972), "Criteria for Protection Systems for Nuclear Power Generating Stations"
2. IEEE 336-1977 (ANSI N45.2), "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
3. IEEE 338-1975, "Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems"
4. IEEE 352-1975 (ANSI N41.4-1976), "General Principles for Reliability Analysis of Nuclear Power Generating Stations"
5. IEEE 380-1975, "Definition of Terms Used in IEEE Nuclear Power Generating Station Standards"
6. IEEE 384-1977 (ANSI N41.14), "Criteria for Independence of Class IE Equipment and Circuits"
7. IEEE 420-1973 (ANSI N41.17), "Class IE Control Switchboards for Nuclear Power Generating Stations"
8. IEEE 494-1974, "Method for Identification of Documents Related to Class IE Equipment and Subsystems for Nuclear Power Generating Stations"
9. IEEE 500-1977, "Guide to the Collection and Presentation of Electrical, Electronic and Sensing Component Reliability Data for Nuclear Power Generating Stations"
10. IEEE 566-1977, "Recommended Practice for Design of Display and Control Facilities for Control Rooms of Nuclear Power Generating Stations"
11. IEEE 577-1976, "Requirements for Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Power Generating Stations"
12. ANSI/ANS 59.3-1977, "Safety Related Control Air Systems"
13. ANSI N544-1968, "Signal Connectors for Nuclear Instruments"

B. RDT Standards

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

1. Cl-1T, "Instrumentation and Control Equipment Grounding and Shielding Practices"
2. Cl6-1T, "Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems"
3. Cl6-2T, "Protection System Logic"
4. Cl6-3T, "PPS Buffers"
5. Cl6-4T, "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 7.2, "Reactor Trip System," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 7.3, "Engineered Safety Feature Systems," *ibid.*
3. USNRC Standard Review Plan 7.4, "Systems Required for Safe Shutdown," *ibid.*

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

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

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

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

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

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

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

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., and 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-3, Draft, "Protection Systems in Nuclear Power Plants," International Atomic Energy Agency, Vienna

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

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

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

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

5. ANS 4.3.3, Proposed, "Criteria for the Application of Digital Computers in Safety Related Functions Other than Protection Systems for Nuclear Power Generating Stations"

7.1.8 Topics for Future Standards Development

None

7.2 NON-SAFETY INSTRUMENTATION AND CONTROL SYSTEMS

7.2.1 Scope

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

7.2.2 Safety Function

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

7.2.3 Definition

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

7.2.4 Design Guidelines

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

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

7.2.4.2 Maintenance, Testing, and Inspection

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

7.2.4.3 Interfaces With Other Systems

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

Figure 7.2-1. Guide to the Use of Section 7.2.

- A. Reactivity Control System
- B. Reactor Coolant System
- C. Reactor Building
- D. Protection System
- E. Non-Class IE Electric Power System
- F. Cooling Water System
- G. Purification and Makeup Water System
- H. Fire Protection System
- I. Experimental Facilities
- J. Radioactive Waste System

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

7.2.5 Applicable General Design Criteria

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

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

7.2.6 Standards, Codes, and Guides

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

- A. Industrial Standards and Codes
 - 1. ANSI C1-1975 (NFPA 70-19075), "National Electrical Code"
 - 2. ANSI C2-1977, "National Electrical Safety Code"
 - 3. ANSI N413-1974, "Guidelines for the Documentation of Digital Computer Programs"
 - 4. IEEE 384-1977 (ANSI N41.14), "Criteria for Independence of Class IE Equipment and Circuits"

5. ISA, "Standards and Practices for Instrumentation," 3rd Edition
6. NEMA IS 1.1, "Enclosures for Industrial Controls"
7. NEMA ICS-1970, "NEMA Standards Publication, Industrial Controls and Systems"
8. ANS STD 3-1971, "Recommended Programming Practices to Facilitate the Interchange of Digital Computer Programs"

B. RDT Standards

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

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

C. Guides

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

7.2.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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., and Beckerly, J.G., "Nuclear Power Reactor Instrumentation System Handbook," Volumes 1 and 2, TID-25952-P1 and -P2, USAEC, 1973 and 1974
3. "Standards and Practices for Instrumentation," Fifth Edition, Instrument Society of America, 1977

B. Pending Industry Standards and Codes

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

7.2.8 Topics for Future Standards Development

None

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CHAPTER 8

ELECTRIC POWER

8.1 CLASS IE ELECTRIC POWER SYSTEMS

8.1.1 Scope

This section provides guidance for determining the need for a Class IE Electric Power System and additionally provides guidelines for the design of this system. Figure 8.1-1 is a guide to the use of this section.

8.1.2 Safety Function

The safety function of the Class IE Electric Power System is to provide sufficient power of suitable quality to safety class systems and equipment in order to assure their ability to perform their safety functions when required.

8.1.3 Definition

The Class IE Electric Power System is the on-site, safety-related electric system and generally includes the following equipment:

- A. AC and/or DC standby power supplies.
- B. The safety-related electric distribution system.
- C. Suitable devices to ensure adequate electrical separation from Non-Class IE portions of the electric power system and non-essential loads.

8.1.4 Design Guidelines

8.1.4.1 Guidelines for Determining the Need for a Class IE Electric Power System

A Class IE Electric Power System should be provided for safety class systems or equipment if electricity must be available for the performance of a required safety function. The systems and equipment that may be supplied with IE power generally include those that are required for safe reactor shutdown, for post-LOCA core cooling and for preventing an uncontrolled release of radioactivity to the environment with dose consequences in excess of the guidelines in Chapter 15.

A Category III reactor facility does not require a Class IE Electric Power System if it exhibits the following general characteristics upon loss of all electric power:

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

Figure 8.1-1. Guide to the Use of Section 8.1.

B. No active, electric-powered systems are required for shutdown cooling of the reactor (including post-LOCA core cooling).

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

A facility with these characteristics may utilize a Non-Class IE Electric Power System (see Section 7.2) for all station electrical needs. This provision may apply to the smaller Category III reactors that do not require an ECCS or a confinement.

8.1.4.2 Guidelines for the Design of a Class IE Electric Power System

A. The Class IE Electric Power System should be designed with suitable redundancy and separation of redundant components in order to ensure that adequate electrical capacity and capability is available during a postulated accident with the addition of a single failure of a component in the Class IE Electric Power System. Application of the single failure criterion to system design is discussed in Chapter 15.

B. The Class IE portions of an electric power system should be Seismic Category I and should be qualified as described in Section 3.10.

C. The system should be designed to remain functional under the environmental conditions discussed in Section 3.11 for a period of time sufficient to ensure that all required safety functions are completed.

D. The on-site standby electric power sources, which supply emergency Class IE power, should be suitably separated (e.g., via transformers and circuit breakers) from the off-site power supplies, which normally provide electric power to the facility, and from other Non-Class IE portions of the on-site electric distribution system.

E. Non-Class IE loads which may be supplied from a Class IE bus should be provided with suitable devices to ensure their disconnection during emergencies.

F. The Class IE Electric Power System should be protected from the effects of severe natural phenomena and missiles as described in Chapter 3.

8.1.4.3 Maintenance, Testing, and Inspection

A. The Class IE Electric Power System should be capable of periodic testing to verify system performance.

B. Bypassed and inoperable status indication should be provided to the operator to indicate that portions of the Class IE electric power system have a reduced operational capability due to maintenance or testing in progress.

8.1.4.4 Interfaces With Other Systems

The Class IE Electric Power System may interface with the following other facility systems:

- A. Emergency Core Cooling System
- B. Reactor Building
- C. ESF Ventilation System
- D. Protection System

The design of the Class IE Electric Power System should ensure a compatible application of codes and standards across all interfaces with other facility systems.

8.1.5 Applicable General Design Criteria

In addition to the 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 Class IE Electric Power Systems:

- A. GDC 17, "Electric Power Systems"
- B. GDC 18, "Inspection and Testing of Electrical Power Systems"
- 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 29, "Protection Against Anticipated Operational Occurrences"
- I. 10 CFR 50.55a, "Codes and Standards"

8.1.6 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 Class IE Electric Power Systems:

- A. Industry Standards and Codes

- 1. IEEE 279-1971 (ANSI N42.7-1972), "Criteria for Protection Systems for Nuclear Power Generating Stations"

2. IEEE 308-1974 (ANSI N41.12-1975), "Class IE Power Systems for Nuclear Power Generating Stations"

3. IEEE 336-1971 (ANSI N45.2-1972), "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"

4. IEEE 338-1975, "Criteria for Periodic Testing of Nuclear Power Generating Station Protection Systems"

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

6. IEEE 384-1974 (ANSI N41.14), "Criteria for Separation of Class IE Equipment and Circuits"

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

8. IEEE 446-1974, "Recommended Practice for Emergency and Standby Power Systems"

9. IEEE 450-1975 (ANSI N41.15-1976), "Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations"

10. IEEE 484-1975 (ANSI N41.24-1976), "Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations"

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

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

13. IEEE 505-1977, "Nomenclature for Generating Station Electric Power Systems"

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

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.6, "Independence Between Redundant Standby (On-Site) Power Sources and Between Their Distribution Systems"

2. USNRC Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions"

3. USNRC Regulatory Guide 1.32, "Use of IEEE 308-1971, Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"

4. USNRC Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Protection Systems"

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

6. USNRC Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants"

7. USNRC Regulatory Guide 1.93, "Availability of Electric Power Sources"

8. USNRC Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"

9. USNRC Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection System"

10. USNRC Regulatory Guide 1.128, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants"

11. USNRC Regulatory Guide 1.129, "Maintenance, Testing Replacement of Large Lead Storage Batteries for Nuclear Power Plants"

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

8.1.7 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 8.2, "Off-Site Power System," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 8.3.1, "AC Power Systems (Onsite),"
ibid.

3. USNRC Standard Review Plan 8.3.2, "DC Power Systems (Onsite),"
ibid.

4. Branch Technical Position EICSB 6, "Capacity Test Requirements of Station Batteries--Technical Specifications," ibid.

5. Branch Technical Position EICSB 7, "Shared Onsite Emergency Electrical Power Systems for Multi-Unit Generating Stations," ibid.

6. Branch Technical Position EICSB 11, "Stability of Offsite Power Systems," ibid.

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

8. Branch Technical Position EICSB 27, "Design Criteria for Thermal Overload Protection for Motors of Motor-Operated Valves," ibid.

9. IAEA Safety Guide SG-D7, Draft, "Safety Related Electric Power Systems," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

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

8.1.8 Topics for Future Standards Development

None

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8.2 NON-CLASS IE ELECTRIC POWER SYSTEMS

8.2.1 Scope

This section provides guidance relevant to the design of a Non-Class IE Electric Power System. Figure 8.2-1 is a guide to the use of this section.

8.2.2 Safety Function

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

8.2.3 Definition

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

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 does 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

None

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. Reactor Coolant System
- C. Non-Safety Instrumentation and Control Systems

Figure 8.2-1. Guide to the Use of Section 8.2.

- D. Cooling Water System
- E. Fuel Handling and Storage System
- F. Purification and Makeup Water System
- G. Fire Protection System
- H. Experimental Facilities
- I. Radioactive Waste System
- J. Radiation Protection System

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

8.2.5 Applicable General Design Criteria

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

- A. GDC 17, "Electric Power Systems"
- B. GDC 24, "Separation of Protection and Control Systems"

8.2.6 Standards, Codes, and Guides

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

- A. Industry Standards and Codes
 - 1. ANSI C1-1975 (NFPA 70-1975), "National Electrical Code"
 - 2. ANSI C2-1977, "National Electrical Safety Code"
 - 3. IEEE 27-1969 (ANSI C37.10-1969), "Switchgear Assemblies, Including Metal Enclosed Bus"
 - 4. IEEE 141-1976, "Electric Power Distribution for Industrial Plants"
 - 5. IEEE 142-1972 (ANSI C114.1-1973), "Recommended Practice for Grounding Industrial and Commercial Power Systems"
- B. RDT Standards
 - None

C. Guides

None

8.2.7 Supplementary Information

None

8.2.8 Topics for Future Standards Development

None

CHAPTER 9

AUXILIARY SYSTEMS

9.1 COOLING WATER SYSTEMS

9.1.1 Scope

This section provides guidelines for the design of cooling water systems responsible for transferring heat from structures, systems, and components (including the reactor pool and the spent fuel pool) to an ultimate heat sink which rejects the heat to the environment. Figure 9.1-1 is a guide to the use of this section.

9.1.2 Safety Function

Cooling water systems may be required to permit extended operation of a Category III reactor at high power (e.g., >1MW). However, cooling water systems are not generally required for safety and therefore usually have no safety function.

9.1.3 Definition

The cooling water system includes all components that form the generally closed loop fluid flow paths from the equipment or systems requiring cooling to the ultimate heat sink. These components include pumps, valves, piping, flow control devices, heat exchangers and the ultimate heat sink.

9.1.4 Design Guidelines

9.1.4.1 Guidelines for Determining the Minimum Safety Class of a Cooling Water System

A. The cooling water system does not generally have a safety function; therefore, the system may be non-nuclear.

9.1.4.2 Guidelines for the Design of a Cooling Water System

A. Not all Category III reactor facilities require active cooling of the reactor coolant system. Facilities operated at low power levels or only intermittently at high power levels may be capable of rejecting sufficient heat from the reactor coolant system to passive heat sinks in order to maintain acceptably low reactor coolant bulk temperatures. These facilities do not require a cooling water system to remove reactor coolant system heat. Typically, acceptable upper limits on reactor coolant bulk temperature are in the range of 120 F to 140 F. Facilities that can not adequately reject heat from the reactor coolant system to passive heat sinks should be provided with active reactor coolant system cooling or should be operated under technical specification limits on extended operation at high power.

Figure 9.1-1. Guide to the Use of Section 9.1.

B. When the cooling water system is designed to provide for reactor coolant system cooling, the heat exchangers may be located in the reactor pool to enhance natural circulation of the reactor coolant if the release of gaseous nuclides (e.g., ^{16}N , ^{41}Ar) to the reactor building atmosphere do not result in doses in excess of the guidelines of Chapter 12. If this system configuration would result in exposure in excess of the guidelines in Chapter 12, it is recommended that heat exchangers for the reactor coolant system be located external to the reactor pool. The reactor coolant system then would utilize forced circulation through loop piping that will ensure sufficient decay of the short-lived gaseous nuclides before the return of the reactor coolant to the pool (see Section 5.1.).

C. The cooling water system should have a heat utilization capability at least equal to the maximum heat load imposed under any mode of normal operation or upset conditions (these plant conditions are defined in Table 3.9-1). The maximum heat load may include reactor generated heat, decay heat from spent fuel (if spent fuel is stored in the reactor pool) and heat from other components requiring cooling.

D. A cooling water system that transfers heat from the reactor coolant system or from the spent fuel pool should include provisions for detecting radioactivity in the cooling water system.

E. The cooling water system design should include provisions for isolating leaking heat exchangers or other leaking components.

F. Suitable cooling water system instrumentation should be provided to assess system performance and to provide for necessary system control functions.

G. The cooling water system may be operable on off-site electric power only.

H. Mechanical portions of cooling water systems should consider the design guidance in Section 3.9.

9.1.4.3 Maintenance, Testing, and Inspection

A. The cooling water system should be designed to permit initial and periodic hydrostatic tests to verify system integrity.

B. The cooling water system should be designed to permit periodic tests of system performance.

9.1.4.4 Interfaces With Other Systems

The cooling water system may interface with the following other facility systems:

A. Reactor Coolant System

B. Reactor Building

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

The design of cooling water systems should ensure a compatible application of codes and standards across all interfaces with other facility systems.

9.1.5 Applicable General Design Criteria

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

- A. GDC 44, "Cooling Water"
- B. GDC 45, "Inspection of Cooling Water System"
- C. GDC 46, "Testing of Cooling Water System"

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 standards, codes, and guides be considered in the design of cooling water systems:

A. Industry Standards and Codes

- 1. ANSI N18.5 (ANS 5.1), Draft, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors"

B. RDT Standards

None

C. Guides

- 1. USNRC Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"

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 9.2.1, "Station Service Water System," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 9.2.2, "Reactor Auxiliary Cooling Water Systems," *ibid.*
3. USNRC Standard Review Plan 9.2.5, "Ultimate Heat Sink," *ibid.*
4. IAEA Safety Guide SG-D6, Draft, "Ultimate Heat Sink," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

1. ANSI N188, Draft, "Safety Related Cooling Water Systems in Nuclear Power Plants"

9.1.8 Topics for Future Standards Development

None

9.2 FUEL HANDLING AND STORAGE SYSTEM

9.2.1 Scope

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

9.2.2 Safety Function

The fuel handling and storage system should provide for handling and storage of new and irradiated fuel assemblies in a manner which ensures:

- A. No inadvertent criticality during fuel handling or storage.
- B. Adequate spent fuel decay heat removal.
- C. Adequate radiation protection for fuel handling personnel.

9.2.3 Definition

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

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

9.2.4 Design Guidelines

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

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

- 1. Their failure may result in potential off-site doses in excess of 0.5 rem.

Figure 9.2-1. Guide to the Use of Section 9.2.

2. They are required for adequate spent fuel decay heat removal.

3. They are required for maintaining new or spent fuel in a subcritical configuration.

B. Portions of the fuel storage facility located within the reactor pool should be designed to a safety class that is compatible with the reactor coolant system.

C. Portions of the fuel handling and storage system that pass through a safety class structure (e.g., reactor building) should be designed to a safety class that is compatible with the structure.

D. Other portions of the fuel handling and storage system may be nonnuclear.

9.2.4.2 Guidelines for the Design of Fuel Handling and Storage Systems

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

B. The spent fuel storage facility should be designed to ensure that the dose guidelines presented in Chapter 15 are not exceeded during any facility operating condition. During normal operating conditions, the spent fuel storage facility may include fuel elements with intact or damaged cladding (damaged cladding is defined in Section 4.1). The damaged cladding may be the result of the small portion of fuel element cladding that fails during normal operation or, it may be the result of damage during an accident while in the reactor. Any subsequent damage to the fuel due to accidents while in the spent fuel storage facility (e.g., due to loss of spent fuel cooling or loss of spent fuel water) should be limited such that the dose guidelines for the appropriate facility operating condition are not exceeded.

Based on the water loss tests on the ASTR and LITR reactors discussed in Sections 5.1 and 6.1, loss of water from the spent fuel storage pool (or the reactor pool, when it is used for spent fuel storage) of most Category III reactors should generally not cause significant fuel or cladding damage to the spent fuel because:

1. The operating parameters of the Category III reactor prior to shutdown and removal of the spent fuel were generally below the LITR values of power level, average thermal flux and power density discussed in Sections 5.1 and 6.1, thereby limiting the initial decay heat generation rate of the spent fuel assemblies.

2. The time period from reactor shutdown until removal and storage of spent fuel elements allows for a significant reduction in the spent fuel decay heat generation rate due to fission product decay.

3. Low spent fuel temperature prior to coolant loss. The spent fuel cladding would, thus, be expected to remain an effective physical barrier against the release of fission products from the spent fuel following a loss of spent fuel coolant.

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

D. Safety class portions of the fuel handling and storage system should be Seismic Category I.

E. New and spent fuel storage facilities should be designed such that a fuel assembly cannot be placed anywhere in the storage racks other than the design locations.

F. New fuel storage vaults should include provisions for drainage.

G. Drains, permanently connected systems and other features that by maloperation or failure would cause loss of coolant that would uncover the spent fuel should not be included in the design of spent fuel storage facilities if the loss of coolant would cause fuel damage in excess of the guidelines described in B. above.

H. The spent fuel storage facility should be provided with a makeup water system (see Section 9.3) to maintain the required coolant inventory for adequate decay heat removal such that the spent fuel damage guidelines described in B. above are not exceeded.

I. The spent fuel storage facility should be provided with a cleanup or purification system to remove dissolved and particulate radionuclides from the pool water and to maintain the desired coolant chemistry (see Section 9.3).

J. The fuel handling and storage system should be adequately shielded to ensure that occupational exposures associated with fuel handling are maintained as low as reasonably achievable (see Chapter 12).

K. The fuel handling and storage system should be designed to protect new and spent fuel from the effects of severe natural phenomena and missiles and from mechanical damage from other sources (e.g., items dropped into the spent fuel storage pool).

L. Fuel handling equipment should be designed to prevent dropping a fuel assembly during handling.

M. The reactor facility and the fuel handling system should be designed to prevent damage to fuel assemblies or safety class systems if a heavy load (e.g., a spent fuel cask) is dropped. The facility design may afford this protection by providing physical separation or protection of the fuel by suitable structures. An alternative is to design the handling system such that a connected load will not drop in the event of a single failure or malfunction (e.g., handling system design incorporating multiple, independent load paths).

N. 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.

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

P. Fuel storage facilities located within the reactor pool should be designed such that potential interactions between the stored fuel and the reactor core are negligible.

9.2.4.3 Maintenance, Testing, and Inspection

Safety class fuel handling system components and equipment should be designed to facilitate periodic testing and inservice inspection.

9.2.4.4 Interfaces With Other Systems

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

- A. Reactor Core Assembly
- B. Reactivity Control System
- C. Reactor Coolant System
- D. Reactor Building
- E. Engineered Safety Feature Ventilation System
- F. Non-Safety I&C System
- G. Non-Class IE Electric Power System
- H. Cooling Water System
- I. Purification and Makeup Water System
- J. Fire Protection 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.

9.2.5 Applicable General Design Criteria

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

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

9.2.6 Standards, Codes, and Guides

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

- A. Industry Codes and Standards
 - 1. ANSI B96.1-1973, "Specification for Welded Aluminum-Alloy Field Erected Storage Tanks"
 - 2. ANSI N16.1-1975, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"
 - 3. ANSI N16.5-1975, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials"
 - 4. ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"
 - 5. ANSI N18.6 (ANS 5.1), Draft, "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors"
 - 6. ANSI N210-1976, "Design Objectives for LWR Spent Fuel Facilities"
 - 7. API 620, "Recommended Rules for Design and Construction of Large, Welded Low-Pressure Storage Tanks," American Petroleum Institute, 1973
 - 8. API 650, "Welded Steel for Oil Storage," American Petroleum Institute, 1973
 - 9. AWWA D100-73, "Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage," American Water Works Association
 - 10. CMAA Specification No. 70, "Specification for Electric Overhead Traveling Cranes," Crane Manufacturers Association of America

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis"
2. USNRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"
3. USNRC Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants"
4. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"
5. USNRC Regulatory Guide 3.41, "Validation of Computational Methods for Nuclear Criticality Safety"
6. USNRC Regulatory Guide 3.43, "Nuclear Criticality Safety in the Storage of Fissile Materials"

9.2.7 Supplementary Information

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

A. Background

1. USNRC Standard Review Plan 9.1.1, "New Fuel Storage," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 9.1.2, "Spent Fuel Storage," *ibid.*
3. USNRC Standard Review Plan 9.1.3, "Spent Fuel Pool Cooling and Cleanup Systems," *ibid.*
4. USNRC Standard Review Plan 9.1.4, "Fuel Handling System," *ibid.*
5. USNRC Standard Review Plan 9.4.2, "Spent Fuel Pool Area Ventilation System," *ibid.*
6. Branch Technical Position APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants," (attached to SRP 9.1.4), *ibid.*

B. Pending Industry Standards and Codes

1. ANSI N208 (ANS 57.1), Proposed, "LWR Fuel Handling System Design".
2. ANSI N209 (ANS 57.3), Proposed, "LWR New Fuel Facilities"

9.2.8 Topics for Future Standards Development

None

9.3 PURIFICATION AND MAKEUP WATER SYSTEMS

9.3.1 Scope

This section provides guidelines for the design of a purification system that maintains the coolant chemical composition and activity level of the reactor coolant water and spent fuel pool water within acceptable levels. In addition, guidelines are provided for the design of a makeup water system that maintains the desired coolant inventory in the reactor pool and the spent fuel storage pool. Figure 9.3-1 is a guide to the use of this section.

9.3.2 Safety Function

The safety function of makeup water systems is to maintain sufficient coolant inventory such that:

A. Small breaks in the reactor coolant system do not result in fuel damage in excess of the guidelines of Section 4.1.

B. Water loss from the spent fuel pool does not result in spent fuel damage in excess of the guidelines of Section 9.2.

The purification system generally does not have a safety function.

9.3.3 Definition

Purification and makeup water systems generally include the following major fluid system components:

A. Purification system, including pumps, piping, valves, filters, ion-exchange resin beds, facilities for chemical addition and facilities for obtaining a sample of coolant for chemical and radiochemical analysis.

B. Makeup water system, including piping, valves, charging pumps and water storage tanks required for maintaining the desired coolant inventory.

9.3.4 Design Guidelines

9.3.4.1 Guidelines for Determining the Minimum Safety Class of the Purification and Makeup Water System

A. Portions of the makeup water system should be designed to a minimum of Safety Class 3 if required for:

1. Preventing fuel damage in excess of the guidelines of Section 4.1 as a result of a small break in the reactor coolant system boundary.

2. Preventing spent fuel damage in excess of the guidelines in Section 9.2 as a result of coolant loss from the spent fuel pool.

3. Preventing off-site doses in excess of 0.5 rem.

Figure 9.3-1. Guide to the Use of Section 9.3.

Figure 9.3-1. Guide to the Use of Section 9.3
(Continued).

B. Other portions of the makeup water system may be nonnuclear.

C. The purification system may generally be designed to non-nuclear standards.

9.3.4.2 Guidelines for the Design of a Purification and Makeup Water System

A. The makeup water system for the reactor coolant system should generally be sized to prevent fuel damage in excess of the guidelines in Section 4.1 as a result of leakage from a small break in the reactor coolant system boundary. For reactors which may not incur significant fuel damage due to such leakage, the makeup water system capacity may be based on other considerations such as maintaining adequate water for radiation shielding or for cooling spent fuel stored in the reactor pool (see Section 9.2). Herein, a small break in the reactor coolant system is defined as a breach of small piping or other small components that are part of or attached to the reactor coolant system boundary (e.g., instrument lines).

B. For reactors with a spent fuel storage facility that is not located in the reactor pool, the makeup water system for the spent fuel storage pool should generally be sized to prevent spent fuel damage in excess of the guidelines of Section 9.2. For reactor facilities which may not incur significant damage to spent fuel due to loss of water from the spent fuel storage pool, the makeup water system capacity may be based on other considerations such as maintaining adequate water for radiation shielding.

C. Safety class makeup water systems should be Seismic Category I and should include an appropriately sized Seismic Category I water supply. Non-safety class makeup water systems may utilize any suitable water supplies.

D. A safety class makeup water system should be operable on off-site and on-site Class IE power (assuming one is unavailable). The purification system and non-safety class portions of the makeup water system need only be operable on off-site power.

E. The purification system should be capable of maintaining the amount of dissolved and particulate radionuclides in the reactor coolant (and the spent fuel storage pool water, if a separate storage pool is provided) at low levels such that occupational doses are maintained as low as reasonably achievable (see Chapter 12).

F. Purification system filters, ion-exchangers and other potential radiation sources in the system should be adequately shielded (see Chapter 12).

G. Mechanical portions of the purification and makeup water system should consider the design guidance in Section 3.9.

9.3.4.3 Maintenance, Testing, and Inspection

A. The purification system should be designed to facilitate obtaining suitable water samples in order to determine system effectiveness and to determine the need for filter and ion-exchanger replacement or rejuvenation.

B. Safety class portions of a makeup water system should be designed to permit periodic operational testing and inspection to verify system performance.

9.3.4.4 Interfaces With Other Systems

The purification and makeup water systems may interface with the following other reactor facility systems:

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

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

9.3.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDC be considered in the design of safety and non-safety class components of the purification and makeup water system:

- A. GDC 33, "Reactor Coolant Makeup"

9.3.6 Standards, Codes, and Guides

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

- A. Industry Codes and Standards

1. ANSI B96.1-1973, "Specification for Welded Aluminum-Alloy Field Erected Storage Tanks"

2. API 620, "Recommended Rules for Design and Construction of Large, Welded Low-Pressure Storage Tanks," American Petroleum Institute, 1973

3. API 650, "Welded Steel Tanks for Oil Storage," American Petroleum Institute, 1973

4. AWWA D100-1973, "Standard for Steel Tanks, Stand Pipes, Reservoirs and Elevated Tanks for Water Storage," American Water Works Association

B. RDT Standards

None

C. Guides

None

9.3.7 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 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 9.2.3, "Demineralized Water Makeup System," *ibid.*

B. Pending Industry Standards and Codes

None

9.3.8 Topics for Future Standards Development

None

9.4 FIRE PROTECTION SYSTEM

9.4.1 Scope

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

9.4.2 Safety Function

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

9.4.3 Definition

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

9.4.4 Design Guidelines

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

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

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

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

9.4.4.2 Guidelines for the Design of a Fire Protection System

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

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

Figure 9.4-1. Guide to the Use of Section 9.4.

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

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

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

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

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

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

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

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. Use of water to fight fires in the reactor building or a new fuel storage area is closely supervised by qualified individuals in order to prevent disabling engineered safety feature equipment or causing an inadvertent criticality of new fuel in dry storage.

9.4.4.3 Maintenance, Testing, and Inspection

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

9.4.4.4 Interfaces With Other Systems

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

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

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

9.4.5 Applicable General Design Criteria

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

- A. GDC 3, "Fire Protection"

9.4.6 Standards, Codes, and Guides

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

- A. Industry Codes and Standards

- 1. Fire Protection Equipment

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

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

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

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

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

- f. ANSI A54.5-1974 (NFPA 12B-1973), "Halogenated Fire Extinguishing Agent Systems - Halon 1211"
- g. ANSI Z112.1-1974 (NFPA 10-1974), "Installation, Maintenance and Use of Portable Fire Extinguishers"
- h. ANSI Z273.1-1974 (NFPA 14-1974), "Installation of Standpipe and Hose Systems"
- i. ANSI Z274.1-1974 (NFPA 16-1974), "Installation of Foam-Water Sprinkler Systems and Foam-Water Spray Systems"
- j. ANSI Z275.1-1974 (NFPA 17-1973), "Dry Chemical Extinguishing Systems"
- k. ANSI Z285.1-1974 (NFPA 69-1973), "Explosion Prevention Systems"
- l. ANSI Z286.1-1974 (NFPA 11-1973), "Foam Extinguishing Systems"
- m. ANSI Z286.2-1974 (NFPA 11A-1970), "High Expansion Foam Systems"
- n. ANSI Z286.3-1974 (NFPA 11B-1973), "Synthetic Foam and Combined Agent Systems"
- o. NFPA 13A-1971, "Maintenance of Sprinkler Systems"
- p. NFPA 20-1974, "Standard for the Installation of Centrifugal Fire Pumps"
- q. NFPA 22-1974, "Water Tanks"
- r. NFPA 70B-1975, "Electrical Equipment Maintenance"
- s. NFPA 72-1975, "Standard for the Installation, Maintenance and Use of Proprietary Protective Signaling Systems"
- t. NFPA 72E-1974, "Automatic Fire Detectors"
- u. NFPA 75-1972, "Electronic Computer/Data Processing Equipment"
- v. NFPA 80-1975, "Fire Doors and Windows"
- w. NFPA 90A-1975, "Air Conditioning and Ventilating Systems"
- 2. Fire Protection Administration
 - a. NFPA 6-1974, "Industrial Fire Loss Prevention"
 - b. NFPA 7-1974, "Fire Emergencies Management"

- c. NFPA 8-1974, "Effects of Fire on Operations Management Responsibility"
- d. NFPA 24-1973, "Standard for Outside Protection"
- e. NFPA 26-1958, "Supervision of Valves"
- f. NFPA 27-1975, "Private Fire Brigade"
- g. NFPA 30-1973, "Flammable and Combustible Liquids Code"
- h. NFPA 51B-1971, "Cutting and Welding Processes"
- i. NFPA 241-1975, "Safeguarding Buildings During Construction and Demolition Operations"
- j. NFPA 251-1972 (ASTM E 119-1971), "Standard Methods of Fire Tests of Building Construction and Materials"
- k. NFPA 255-1972 (ASTM E 84-1971), "Method of Test of Surface Burning Characteristics of Building Material"
- l. NFPA 601-1975, "Guard Service in Fire Loss Prevention"
- m. NFPA 801-1975, "Recommended Fire Protection Practices for Facilities Handling Radioactive Materials"
- n. NFPA 802-1974, "Recommended Fire Protection Practice for Nuclear Reactors"

B. RDT Standards

None

C. Guides

1. Fire Protection System Equipment

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

2. Fire Protection Administration

- a. USNRC Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants"
- b. USNRC Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants"

9.4.7 Supplementary Information

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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," Rev. 1, NUREG 75/087, September 1975
2. Branch Technical Position APCSB 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants," (attached to SRP 9.5.1, Rev. 1)
3. NUREG-0050, "Recommendations Related to Browns Ferry Fire," USNRC, February 1976
4. IAEA Safety Guide SG-D2, Draft, "Safety Guide for Fire Protection in Nuclear Power Plants," International Atomic Energy Agency, Vienna
5. WASH 1245-1, "Standard for Fire Protection of AEC Electric Computer/Data Processing Systems"

B. Pending Industry Standards and Codes

1. ANS 15.17, Proposed, "Fire Protection for Research Reactors"
2. ANSI N18.10/ANSS 59.4, Draft, "Generic Requirements for Light Water Nuclear Power Plant Fire Protection"
3. IEEE P634, Draft, "Standard for Cable Penetration Fire Stop Qualification Test"

9.4.8 Topics for Future Standards Development

None

CHAPTER 10

EXPERIMENTAL FACILITIES

10.1 SCOPE

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

10.2 SAFETY FUNCTION

The safety functions of the experimental facilities are those associated with potential interfaces with the reactor core assembly (see Section 4.1), the reactivity control system (see Section 4.2), the reactor coolant system (see Section 5.1) or other safety class systems.

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

10.3 DEFINITION

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

10.4 DESIGN GUIDELINES

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

A. In-core experimental facilities (e.g., irradiation holes) should be designed to a safety class that is compatible with the safety requirements of the reactor core and the safety class of the reactor pool.

B. If the failure of an in-core experimental facility does not prevent safe shutdown and cooling of the reactor, but may result in potential off-site doses in excess of the 10 CFR 100 guidelines, the experimental facilities should be designed to a minimum safety class of SC-2.

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

Figure 10.1-1. Guide to the Use of Chapter 10.

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

10.4.2 Guidelines for the Design of Experimental Facilities

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

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

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

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

1. Streaming from radiography facilities
2. Scattering from ex-core components
3. Fissionable experiments
4. Irradiated sample handling
5. Activated portions of experimental facilities
6. Other sources

Guidelines for occupational radiation exposures are included in Chapter 12.

E. Mechanical portions of experimental facilities should consider the guidelines in Section 3.9.

F. The instrumentation and controls associated with the experimental facility should include appropriate interlocks if operation of the experimental facility can create conditions that may adversely affect the safe operation of the reactor. If interlock circuits interface with the protection system, the guidelines in Section 7.1 should be considered in their design.

G. Strict review procedures for experiments and experiment plans should be established to ensure that proposed experiments satisfy the following general guidelines:

1. The proposed experiments should fall within limits for experiments delineated in the reactor facility technical specifications.

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

3. The proposed experiment should not present an undue hazard to health and safety (see radiological consequence guidelines in Chapter 15).

4. The proposed experiment should not present an undue hazard to the reactor facility.

10.4.3 Maintenance, Testing, and Inspection

A. Prior to loading in the reactor core, experiments (particularly fissionable or explosive experiments) should be qualified for the expected environment by test, analysis, operating experience, or a combination of these methods. Testing of actual equipment under conditions which simulate the expected service conditions, to the extent practicable, (with the exception of the radiation environment) is the preferred method.

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

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

10.4.4 Interfaces With Other Systems

Experimental facilities may interface with the following other facility systems:

- A. Reactor Core Assembly
- B. Reactivity Control System
- C. Reactor Coolant 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 GENERAL DESIGN CRITERIA

No General Design Criteria are specifically applicable to the design of experimental facilities. The GDCs that are applicable to interfacing systems should be considered in the design of experimental facilities that may potentially affect the safe operation of the reactor facility. In addition, the following 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 N5.2-1963, "Design Guide for a Radioisotope Laboratory"
2. ANSI N16.1-1975, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"
3. ANSI N399-1974, "Records and Reports for Research Reactors"
4. ANSI N401-1974, "Review of Experiments for Research Reactors"
5. ANSI N405-1975, "Safety Guide for the Performance of Critical Experiments"

B. RDT Standards

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

B. Pending Industry Standards and Codes

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

10.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 11

RADIOACTIVE WASTE SYSTEMS

The ERDA Manual Chapter 0511, "Radioactive Waste Management," provides guidelines for establishing a program to manage radioactive waste in such a manner as to minimize the potential radiation exposure and associated risk to man. Chapter 0513 of the ERDA Manual, "Effluent and Environmental Monitoring and Reporting," prescribes monitoring requirements that are applicable to systems used to monitor radioactive waste areas. Guidelines for exposure to radiation and for maximum concentrations of radioactive material in air and water effluents are included in the ERDA Manual Chapter 0524. These guidelines establish functional requirements for radioactive waste system designs. Radioactive waste material to be transported from a facility should be prepared for shipment in accordance with the provisions of the ERDA Manual Chapter 0529, "Safety Standards for the Packaging of Fissile Material and Other Radioactive Material."

This chapter provides guidelines for implementing a radioactive waste management program in the design phase of facility modifications or in the design of new facilities. Figure 11.1-1 is a guide to the use of this chapter.

11.2 SAFETY FUNCTION

The safety function of the radioactive waste system is to collect and process radioactive waste material in such a manner that exposure to on-site and off-site personnel is maintained within the guidelines for the appropriate facility operating condition (see Chapter 15).

11.3 DEFINITION

The radioactive waste system includes the piping, valves, tanks 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 condition (see Chapter 15).

11.4 DESIGN GUIDELINES

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

A. Portions of the radioactive waste system should be designed to a minimum safety class of SC-3 if their failure may result in potential off-site doses in excess of 0.5 rem.

Figure 11.1-1. Guide to the Use of Chapter 11.

B. Other portions of the radioactive waste system may be nonnuclear.

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. Reactor coolant purification.
2. Confinement ventilation system filters
3. Fuel handling and storage system
4. Contaminated components, or equipment used for maintenance, repair, sampling or testing.
5. Cleaning solvents and rags.
6. Reactor building sumps.
7. Waste from radiochemical analysis.
8. Experimental facilities.

B. The radioactive waste system should be designed such that exposures resulting from releases during normal operations are maintained as low as reasonably achievable below the ERDA Manual Chapter 0524 guidelines (see Chapter 12).

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

D. Safety class radioactive waste system components should be protected against the effects of severe natural phenomena and missiles (see Chapter 3).

E. Safety class radioactive waste system components should be Seismic Category I.

F. The capability for reducing the volume of radioactive waste should be provided where practical.

11.4.3 Maintenance, Testing, and Inspection

Monitoring and control equipment should include provisions for periodic testing, inspection, and calibration.

11.4.4 Interfaces With Other Systems

The radioactive waste system may interface with the following other facility systems:

- A. Reactor Coolant 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. Purification and Makeup Water System
- H. Fire Protection System
- I. Experimental Facilities
- J. Radiation Protection System

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

11.5 APPLICABLE GENERAL DESIGN CRITERIA

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, it is recommended that the following GDCs and other provisions of 10 CFR be considered in the design of safety class or non-nuclear portions of radioactive waste system:

- A. GDC 60, "Control of Releases of Radioactive Materials to the Environment"
- B. GDC 61, "Fuel Storage and Handling and Radioactivity Control"
- C. GDC 63, "Monitoring Fuel and Waste Storage"
- D. GDC 64, "Monitoring Radioactivity Releases"
- E. 10 CFR 20, "Standards for Protection Against Radiation"
- F. 10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Materials in Effluents - Nuclear Power Plants"
- G. 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Plants"
- H. 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear 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 Effluents of Light Water-Cooled Power Plants"
2. USNRC Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light Water-Cooled Nuclear Power Reactors"
3. USNRC Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors"

11.7 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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," NUREG 75/087, September 1975
2. WASH-1202, "Plan for the Management of AEC-Generated Radioactive Wastes"

B. Pending Industry Standards and Codes

1. ANSI N498 (ANS 40.12), Proposed, "General Criteria for Waste Management"
2. ANSI N525 (ANS 40.11, Proposed, "Definition of Radioactive Waste Categories"

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

11.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 12

RADIATION PROTECTION

12.1 SCOPE

Requirements for radiation protection are specified in the following DOE Manual chapters:

- A. 0502, "Notification, Investigation and Reporting of Occurrences"
- B. 0513, "Effluent and Environmental Monitoring and Reporting"
- C. 0524, "Standards for Radiation Protection"
- D. 0525, "Occupational Radiation Exposure Information"
- E. 0545, "Nuclear Accident Dosimetry Program"

This chapter provides additional design guidelines for radiation protection during normal facility operations, and includes references to detailed design standards that are applicable to radiation protection systems and procedures. Radiological consequence guidelines during postulated accidents are included in Chapter 15. Figure 12.1-1 is a guide to the use of this chapter.

12.2 SAFETY FUNCTION

The safety function of 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 radiation protection system includes the components and systems required for monitoring, surveying, sampling and testing the radiation environment of a facility and its effluents. The system generally includes provisions for alarming abnormal conditions. The radiation protection system hardware also includes permanent and temporary shielding installed to minimize personnel exposure to direct and scattered radiation.

The administrative portion of the radiation protection system includes a system of records, policies and operating procedures which facilitate maintaining occupational and off-site exposures to radiation and radioactive material ALARA.

Figure 12.1-1. Guide to the Use of Chapter 12.

12.4 DESIGN GUIDELINES

12.4.1 Guidelines for the Design of a Radiation Protection System

A. The radiation protection system, incorporating administrative procedures and protective hardware, should be designed to assure that the exposure of personnel to radiation is maintained within the following guidelines:

1. Occupational Radiation Exposures Guidelines

Exposure of personnel in controlled areas should be maintained ALARA in accordance with the ERDA Manual Chapter 0524, Part V and USNRC Regulatory Guide 8.8.

2. Off-Site Radiation Exposure Guidelines

Exposure of persons in unrestricted areas to routine releases of radioactive material should be maintained ALARA in accordance with the ERDA Manual Chapter 0524, Part V and 10 CFR 50, Appendix I.

B. The facility design should ensure adequate protection from direct and scattered radiation and airborne sources. Methods of protection to be considered in facility design include the following:

1. Permanent or temporary shielding.
2. Reactor Building Ventilation Systems (see Section 6.2).
3. 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 protection system actions and operation of engineered safety feature systems should consider the design guidelines in Section 7.1.

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

1. Reactor building radiation and airborne activity levels.
2. Cooling water system activity level (may result from reactor coolant system leakage into cooling water loop).
3. Radioactive waste system areas.

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

12.4.2 Maintenance, Testing, and Inspection

Monitoring equipment should include provisions for periodic testing, inspection, and calibration.

12.4.3 Interfaces With Other Systems

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

- A. Reactor Building
- B. Protection System
- C. Cooling Water System
- D. Fuel Handling and Storage System
- E. Experimental Facilities
- F. Radioactive Waste Systems

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

12.5 APPLICABLE GENERAL DESIGN CRITERIA

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

- A. GDC 60, "Control of Releases of Radioactive Materials to the Environment"
- B. GDC 63, "Monitoring Fuel and Waste Storage"
- C. GDC 64, "Monitoring Radioactivity Releases"
- D. 10 CFR 19, "Notices, Instructions and Reports to Workers; Inspections"

E. 10 CFR 20, "Standards for Protection Against Radiation"

F. 10 CFR 30, "Rules of General Applicability to Licensing of Byproduct Material"

G. 10 CFR 40, "Licensing of Source Material"

H. 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light Water-Cooled Nuclear Power Reactors Effluents"

I. 10 CFR 70, "Special Nuclear Material"

J. 10 CFR 100, "Reactor Site Criteria"

12.6 STANDARDS, CODES, AND GUIDES

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

A. Industry Standards and Codes

1. Radiation Protection System Equipment

a. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Material of Nuclear Facilities"

b. ANSI N13.3-1969, "Dosimetry for Criticality Accidents"

c. ANSI N13.4-1971, "Specification for Portable X- and Gamma-Radiation Survey Instruments"

d. ANSI N13.5-1972, "Performance Specification for Direct Reading and Indirect Readings Pocket Dosimeters for X- and Gamma-Radiation"

e. ANSI N13.7-1972, "Criteria for Film Badge Performance"

f. ANSI N13.10-1974, "On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents"

g. ANSI N16.2-1969, "Criticality Accident Alarm System"

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

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

j. ANSI N322-1977, "Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters"

- k. ANSI N512-1974, "Protective Coatings (Paint) for the Nuclear Industry"
- l. ANSI N648-1975, "Specification for Aggregates for Radiation Shielding Concrete"
- m. ANSI N649-1975, "Descriptive Nomenclature of Constituents of Aggregates for Radiation Shielding Concrete"
- n. ANSI/ANS 6.4-1977 (N403), "Guidelines for the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants"
- 2. Radiation Protection System Administration
 - a. ANSI N2.1-1969, "Radiation Symbol"
 - b. ANSI N2.3-1967, "Immediate Evacuation Signal for Use in Industrial Installations Where Radiation Exposure May Occur"
 - c. ANSI N12.1-1971, "Fissile Material Symbol"
 - d. ANSI N13.2-1969, "Administrative Practices in Radiation Monitoring"
 - e. ANSI N13.6-1966, "Practice for Occupational Radiation Exposure Records Systems"
 - f. ANSI N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants"
 - g. ANSI N18.9-1972, "Program for Testing Biological Shielding in Nuclear Power Plants"
 - h. ANSI N105-1973, "Definition of Terms Relating to Dosimetry"
 - i. ANSI N510-1975, "Testing Nuclear Air Cleaning Systems"
 - j. ANSI Z88.1-1969, "Procedures for Respiratory Protection"
 - k. ANSI/ANS 6.1.1-1977 (N666), "Neutron and Gamma-Ray Flux-to-Dose Rate Factors"
 - l. ANSI/ANS 15.11-1977 (N628), "Radiological Control at Research Reactor Facilities"

B. RDT Standards

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

- 1. E9-1T, "HEPA Filters"

C. Guides

1. Radiation Protection System Equipment

- a. USNRC Regulatory Guide 1.52, "Design, Testing and Maintenance for ESF Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants"
- b. USNRC Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants"
- c. USNRC Regulatory Guide 3.9, "Concrete Radiation Shields"
- d. USNRC Regulatory Guide 4.5, "Measurements of Radionuclides in the Environment--Sampling and Analysis of Plutonium in Soil"
- e. USNRC Regulatory Guide 4.6, "Measurements of Radionuclides in the Environment--Strontium-89 and Strontium-90 Analysis"
- f. USNRC Regulatory Guide 8.3, "Film Badge Performance Criteria"
- g. USNRC Regulatory Guide 8.4, "Direct Reading and Indirect Reading Pocket Dosimeters"
- h. USNRC Regulatory Guide 8.12, "Criticality Accident Alarm Systems"
- i. USNRC Regulatory Guide 8.14, "Personnel Neutron Dosimeters"

2. Radiation Protection System Administration

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

- g. USNRC Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing, Appendix I"
- h. USNRC Regulatory Guide 2.1, "Shield Test Program for Evaluation on Installed Biological Shielding in Research and Training Reactors"
- i. USNRC Regulatory Guide 3.2, "Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles"
- j. USNRC Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants"
- k. USNRC Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants"
- l. USNRC Regulatory Guide 8.1, "Radiation Symbol"
- m. USNRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"
- n. USNRC Regulatory Guide 8.5, "Immediate Evacuation Signal"
- o. USNRC Regulatory Guide 8.7, "Occupational Radiation Exposure Records System"
- p. USNRC Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Reasonably Achievable"
- q. USNRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Reasonably Achievable"
- r. USNRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection"
- s. USNRC Regulatory Guide 10.1, "Compilation of Reporting Requirements of Persons Subject to NRC Regulations"

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 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 12.1, "Assuring that Occupational Radiation Exposures are ALARA," *ibid.*
3. USNRC Standard Review Plan 12.2, "Radiation Sources," *ibid.*
4. USNRC Standard Review Plan 12.3, "Radiation Protection Design Features," *ibid.*
5. USNRC Standard Review Plan 12.4, "Dose Assessment," *ibid.*
6. USNRC Standard Review Plan 12.5, "Health Physics Program," *ibid.*
7. Branch Technical Position ETSB 11-2, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Reactor Plants," (attached to SRP 11.3), *ibid.*
8. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962
9. Shaffer, 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. IAEA Safety Guide SG-05, Draft, "Operational Aspects of Radiological Protection," International Atomic Energy Agency, Vienna
16. "Safe Operation of Critical Assemblies and Research Reactors," 1971 Edition, IAEA Safety Series No. 35, International Atomic Energy Agency, Vienna, 1971
17. MIL-F-51068D, "Filter Particulate High Efficiency Fire Resistant," 1974
18. MIL-STD-282, "Filter Units, Protective Clothing, Gas Mask Components and Related Products: Performance Test Methods," 1956

19. NCRP Report No. 8, "Control and Removal of Radioactive Contamination in Laboratories," 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"

B. Pending Industry Standards and Codes

1. ANSI N17.1, Proposed, "Location and Range of Measurement of Continuous Area and Process Radiation Monitoring System for Light Water Reactors"
2. ANSI N323, Draft, "Radiation Protection Instrumentation Test and Calibration"
3. ANSI N328, Draft, "Control of Radioactive Surface Contamination of Materials, Equipment and Facilities to be Released for Uncontrolled Use"
4. ANSI N336 (ANS 18.2.1), Draft, "Methods for Inferring Environmental Doses"
5. ANSI N346 (ANS 6.6), Draft, "Calculation and Measurement of Direct and Scattered Radiation from Nuclear Power Plants"
6. ANSI N404 (ANS 6.5), Proposed, "Glossary of Terms Shielding and Dosimetry"
7. ANSI N410 (ANS 6.1.2), Proposed, "Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations"
8. ANSI N647 (ANS 15.12), Proposed, "Design Objectives for, and Monitoring of, Systems Controlling Research Reactor Effluents"
9. ANSI N651 (ANS 6.7), Proposed, "Radiation Zoning for Design of Nuclear Power Plants"

10. ANSI N702 (ANS 18.1.2), Draft, "Radioactive Materials in Effluents from Light Water Cooled Nuclear Power Plants"
11. ANSI N703 (ANS 18.1.3), Draft, "Monitoring of Radioactive Materials in Effluents from Light Water Cooled Nuclear Power Plants"
12. ANSI/ANS 6.4 (N403), Draft, "Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants"
13. ANS 2.15, Draft, "Guidelines for Analysis of Atmospheric Transport of Routine Release of Nuclear Reactor Effluent"
14. ANS 58.4, Draft, "Criteria for the Radiological Technical Specifications for Nuclear Power Stations"

12.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 13
CONDUCT OF OPERATIONS

13.1 SCOPE

The following ERDA Manual chapters provide guidance for the conduct of facility operations:

- A. 0502, "Notification, Investigation, and Reporting of Occurrences"
- B. 0530, "Nuclear Criticality Safety"
- C. 0531, "Safety of Nonreactor Nuclear Facilities"
- D. 0540, "Safety of AEC-Owned Reactors"
- E. 0550, "Operational Safety Standards"
- F. 0601, "Emergency Preparedness Programs"
- G. Immediate Action Directives 8401-6 and 8401-7

This chapter provides a listing of other standards and guides that are recommended for consideration in the conduct of facility operations.

13.2 APPLICABLE GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR be considered in the conduct of facility operations:

- A. 10 CFR 19, "Notices, Instructions and Reports to Workers; Inspections"
- B. 10 CFR 50.34, "Contents of Applications, Technical Information"
- C. 10 CFR 50.54, "Conditions of Licenses"
- D. 10 CFR 50.59, "Changes, Tests, and Experiments"
- E. 10 CFR 50, Appendix E, "Emergency Plans for Production and Utilization Facilities"
- F. 10 CFR 55, "Operators' License"
- G. 10 CFR 73, "Physical Protection of Plants and Materials"

13.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in the conduct of facility operations:

A. Industry Standards and Codes

1. ANSI N13.2-1969, "Administrative Practices in Radiation Monitoring"
2. ANSI N15.8-1974, "Nuclear Material Control System for Nuclear Power Plant"
3. ANSI N15.26-1976, "Physical Protection of Special Nuclear Material Within a Facility"
4. ANSI N16.1-1969, "Nuclear Criticality Safety Standard for Operations with Fissionable Materials Outside Reactors"
5. ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"
6. ANSI N18.17-1973, "Industrial Security for Nuclear Power Plants"
7. ANSI N399-1974, "Records and Reports for Research Reactors"
8. ANSI N401-1974, "Review of Experiments for Research Reactors"
9. ANSI N546-1976, "Medical Certification and Monitoring for Personnel Requiring Operator Licenses for Nuclear Power Plants"
10. ANSI/ANS 15.4-1977 (N380), "Selection and Training of Personnel for Research Reactors"

B. RDT Standards

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

1. F1-3T, "Preparation of Unusual Occurrence Reports"
2. F3-2T, "Calibration Program Requirements"
3. F4-20T, "Operation and Maintenance Manuals"

C. Guides

1. USNRC Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage"
2. USNRC Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants"

3. USNRC Regulatory Guide 1.114, "Guidance for Being Operator at the Controls of a Nuclear Power Plant"
4. USNRC Regulatory Guide 1.134, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses"
5. USNRC Regulatory Guide 2.4, "Review of Experiments for Research Reactors"
6. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors"
7. USNRC Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials"
8. USNRC Regulatory Guide 5.29, "Nuclear Material Control System for Nuclear Power Plants"
9. USNRC Regulatory Guide 5.54, "Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants"
10. USNRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"
11. 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. "Safe Operation of Critical Assemblies and Research Reactors, 1971 Edition," IAEA Safety Series No. 35, International Atomic Energy Agency, Vienna, 1971
2. "Manual for the Operation of Research Reactions," IAEA Technical Report Series No. 37, International Atomic Energy Agency, Vienna, 1964
3. "Planning for the Handling of Radiation Accidents," IAEA Safety Series No. 32, International Atomic Energy Agency, Vienna, 1969
4. IAEA Safety Guide SG-01, Draft, "Staffing, Recruitment, Training, and Authorization of Operating Personnel," International Atomic Energy Agency, Vienna

5. USNRC Standard Review Plan 13.1.2, "Operating Organization," NUREG 75/087, September 1975

6. USNRC Standard Review Plan 13.1.3, "Qualifications of Nuclear Plant Personnel," ibid.

7. USNRC Standard Review Plan 13.2, "Training," ibid.

8. USNRC Standard Review Plan 13.3, "Emergency Planning," ibid.

9. USNRC Standard Review Plan 13.4, "Review and Audit," ibid.

10. USNRC Standard Review Plan 13.5, "Plant Procedures," ibid.

11. USNRC Standard Review Plan 13.6, "Industrial Security," ibid.

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

13. NUREG-0094, "A Guide to the Licensing of Facility Operators, Including Senior Operators," USNRC, July 1976

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

B. Pending Industry Standards and Codes

1. ANSI N17.2 (ANS 15.16), Draft, "Standard for Emergency Planning for Research Reactors"

2. ANSI N682 (ANS 3.7.1), Proposed, "Facilities for Medical Care On and Off Plant Site"

3. ANSI N684 (ANS 3.7.2), Proposed, "Emergency Control Centers On and Off Plant Site"

4. ANSI N685 (ANS 3.7.3), Proposed, "Guidance on Conducting Emergency Preparedness Drills"

5. ANSI N700 (ANS 15.14, Draft, "Standard for Physical Security of Research Reactors"

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

13.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 14
INITIAL TEST PROGRAM

14.1 SCOPE

10 CFR 50, Appendix B, Section XI establishes the requirements for an initial startup test program to verify that all structures, systems and components will satisfactorily perform their safety functions at a nuclear facility. The responsibilities and authorities for assuring the adequate protection of health and safety during the initial reactor operations are included in the ERDA Manual, Chapter 0540, "Safety of AEC-Owned Reactors."

This chapter provides a listing of other standards and guides that are recommended for consideration in the conduct of an initial test program.

14.2 APPLICABLE GENERAL CRITERIA

It is recommended that the following provisions of 10 CFR be considered in the conduct of the initial test program:

- A. 10 CFR 50.34, "Contents of Applications; Technical Information"
- B. 10 CFR 50, Appendix B, Section XI, "Test Control"

14.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following codes, standards, and guides be considered in the conduct of the initial test program:

A. Industry Standards and Codes

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

2. IEEE 336-1977, (ANSI N45.24), "Installation, Inspection and Testing Requirements for Instrumentation and Electrical Equipment During Construction of Nuclear Power Generating Stations"

B. RDT Standards

None

C. Guides

None

14.4 SUPPLEMENTARY INFORMATION

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

A. Background Information

1. IAEA Safety Guide SG-04, Draft, "Commissioning Procedures," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

1. ANSI N267 (ANS 3.6), Proposed, "Requirements for Preoperational and Startup Testing"

14.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 15

ACCIDENT ANALYSIS

15.1 SCOPE

The ERDA Manual Chapter 0540, "Safety of AEC-Owned Reactors," requires that the safety of each reactor facility be analyzed, evaluated and documented. This chapter provides general guidelines for performing accident analysis and for applying the single failure criterion. This chapter also recommends standards and guides to be considered in the analysis of postulated accidents at a reactor facility.

15.2 GUIDELINES FOR ACCIDENT ANALYSIS

A. A reactor facility should be designed to accommodate postulated accidents (the initiating event from Table 15.2-1 that starts the accident sequence plus any other events or failures that are a direct or consequential result of the initiating event) within the Table 15.2-2 consequence guidelines for the operating condition that corresponds to the initiating event. With the addition of a single failure (discussed in Section 15.3), the reactor facility should generally accommodate the postulated accident plus single failure within the consequence guidelines for the next higher operating condition (e.g., an emergency event plus a single failure may be accommodated within the faulted consequence guidelines). A faulted event plus a single failure should, however, still be accommodated within the consequence guidelines for the faulted operating condition.

If analysis indicates that the facility cannot meet the appropriate consequence guidelines of Table 15.2-2 for all design basis accidents, some system design changes may be warranted.

B. The radiological consequences of the postulated accidents should not exceed the guidelines in Table 15.2-2. The dose guidelines for the faulted facility operating conditions are discussed in detail below.

1. On-Site Dose Guidelines

In determining the on-site dose consequences that may result from postulated design basis accidents, appropriate credit may be taken for the site emergency plan and the ability to transport and evacuate on-site personnel. Considering this ability to manage on-site personnel, it is recommended that the accident dose consequence goals be considerably less than the faulted operating condition dose guidelines in Table 15.2-2 (25 rem total radiation dose to the whole body and 300 rem total radiation dose to the thyroid).

For individuals who constitute the site emergency force, the guidance in ERDAM 0524, (particularly in Part IV, "Guidance for Emergency Exposures During Rescue and Recovery Activities") should be considered in determining the dose

Table 15.2-1. Spectrum of Postulated Accidents to be Considered in the Design of Pool-Type Reactor Facilities

A. 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 and reactor coolant system), including a reactivity control assembly or temporary control device removal error during refueling. 2. Uncontrolled positive reactivity insertion by the reactivity control system at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power). 3. Reactivity control system maloperation (system malfunction or operator error). 4. Inadvertent loading and operation of a fuel assembly in an improper position. 5. Spectrum of reactivity accidents due to operation or failure of experimental facilities (e.g., sample tube flooding).
B. Radioactive Release from Subsystem or Components
<ol style="list-style-type: none"> 1. Fuel element cladding failure 2. Radioactive waste system leak or failure 3. Fuel handling accidents 4. Accident involving fuel stored in the reactor pool 5. Spent fuel cask drop accidents 6. Experimental facility or specimen failures

Table 15.2-1 (continued)

C. Decrease in Reactor Coolant System Flow Rate	
1.	Reactor coolant system blockage due to failure of in-core experiment facility or any other credible cause
2.	Reactor coolant pump failures (pool type reactors that require forced circulation only) <ul style="list-style-type: none"> a. Reactor coolant pump trip b. Reactor coolant pump shaft seizure c. Reactor coolant pump shaft break
D. Decrease in Reactor Coolant Inventory	
1.	Low pool level due to slow leakage
2.	Loss of coolant accident (LOCA) resulting from the spectrum of postulated breaks in the reactor coolant system boundary, including breaks in the pool liner and in reactor coolant system piping.
E. Increase/Decrease in Core Heat Removal by the Pool Cooling Water System (Pool type reactors that require forced circulation, only)	
F. Anticipated Transients Without Scram	
1.	Inadvertent insertion of positive reactivity by the reactivity control system
2.	Loss of electric power
3.	Loss of heat rejection to pool cooling water system (pool type reactors that require forced circulation, only)

Table 15.2-2. Summary of Facility Operating Conditions and 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, refueling	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, partial loss of cooling water flow, 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	$10^{-1} > F \geq 10^{-3}$	Rod withdrawal accident.	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	$10^{-3} > F \geq 10^{-7}$	Loss of coolant accident.	$\leq 10\text{CFR } 100$	≤ 25 rem whole body and ≤ 300 rem thyroid

*The tabulated frequency ranges for each operating condition are illustrative examples only. They are based on the written description in the table, for a facility with an expected lifetime in the 20-40 year range.

**As low as reasonably achievable.

Table 15.2-2. (Continued) Summary of Facility Operating Conditions and the Corresponding Frequency of Occurrence and Consequence Guidelines.

Operating Condition	General Guidelines	Fuel Design Guidelines	Mechanical System Design Guidelines
Normal	No protection system action required	No fuel or cladding damage (see Section 4.1)	Maximum equipment loading within applicable design code allowables (see Section 3.9)
Upset	The facility should be capable of returning to operation without extensive correction action or repair		
	Facility should be capable of returning to operation following corrective action or repair of damage	Any fuel or cladding damage should not prevent reactor shut-down and adequate core cooling (see Section 4.1)	
Faulted	Facility damage may preclude return to operation		

guidelines that are applicable to emergency force members when performing their duties during an emergency in response to a postulated design basis accident.

2. Off-Site Dose Guidelines

a. At the exclusion area boundary (see Section 2.1), an individual 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.

b. At the outer boundary of the low population zone (see Section 2.1), an individual located at any point on the outer boundary who is exposed to the radioactive cloud from the postulated fission product release resulting from the design basis accident (during the entire period of its passage) will not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

3. Dose Equivalence

Following some postulated accidents, the dose to some critical organ due to inhalation or ingestion or 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 failure do not exceed 0.5 rem to the whole body and if the system is not otherwise required to be Safety Class 1, 2 or 3 (see Table 3.2-1). Likewise, a system should be designed to a minimum of Safety Class 3 if the maximum off-site radiological consequences of system failure exceed 0.5 rem to the whole body and the system is not otherwise required to be Safety Class 1 or 2. As described in Section 3.2, Category III pool-type reactors do not generally require components, systems or structures to be designed to Safety Class 1 standards. The final safety classification of a system is obtained by analysis of the design basis accidents. Due to specific features of an individual Category III reactor facility, analysis of the design basis accidents may indicate that some consequences are not within the guidelines of Table 15.2-2. Therefore, some system design changes may be necessary (i.e., improved reactor core design) to ensure that a final safety class equivalent to the recommended minimum safety class will result in consequences within the guidelines of Table 15.2-2. Alternatively, it may be necessary to increase the quality and reliability of the system by designing to a final safety class that is higher than the minimum safety class recommended by this Design Guide.

D. The spectrum of Design Basis Accidents to be considered in the design Category III reactors is listed in Table 15.2-1. Events such as floods, storms, and earthquakes are not explicitly considered in this chapter. Severe natural phenomena (floods, storms, earthquakes) are discussed in Chapter 3. Probability guidelines applicable to the determination of the spectrum of severe natural phenomena to be considered in reactor facility design are described in Table 15.2-2 and in E. below.

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

The descriptions of the facility operating conditions in Table 15.2-2 include very broad guidelines for determining the facility operating condition to which a particular accident should be assigned. 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 reactor facility based on the expected facility operating lifetime and the nature and consequences of accidents postulated for the facility.

This Design Guide identifies General Design Criteria from 10 CFR 50, Appendix A that may be applicable to Category III reactor facility design. These criteria refer to Anticipated Operational Occurrences (AOOs). Appendix A of 10 CFR 50 defines AOOs as events which are expected to occur one or more times in the life of the reactor facility. Therefore, AOOs include normal, upset and the more likely emergency events. The relationship between AOOs and these facility operating conditions is illustrated in Figure 15.2-1.

E. The spectrum of design basis accidents and severe natural phenomena considered in facility design is generally 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 an accurate quantitative calculation of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of potential doses comparable to the faulted operating condition guidelines is approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative agreements, 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.

15-3(a)

Note: These frequency ranges for each operating condition and for AOOs are illustrative examples only. They are based on the information in Table 15.2-2.

Figure 15.2-1. Approximate Relationship Between the Facility Operating Conditions and the Anticipated Operational Occurrences.

III

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

1. The short term diffusion estimates used for calculating the radiological consequences of postulated accidents should consider the guidance in USNRC Standard Review Plan 2.3.4 regarding the acceptability of models and conservatisms appropriate to the design basis calculations.

2. The maximum steady state reactor power level assumed in the accidents 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 class systems and to the analysis of the consequences of postulated accidents of water-cooled power reactors, the General Design Criteria requires that systems such as the protection system, electric power system, and other safety class systems be designed such that the system safety function can be accomplished with a single failure in the safety class system. This is the basic single failure criterion. Its application to system design assists in determining the minimum system redundancy required to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. An objective in the design of a safety class system is to ensure that a single failure in a component or channel will not interfere with the proper operation of its redundant counterpart.

As described in previous chapters, Category III reactors have very different system requirements than commercial power reactors. Category III reactors have fewer safety class systems, and those which are provided are functionally simpler than those in power reactors. This is due largely to the inherent safety features incorporated in Category III reactor design. The safety systems that are provided, however, should still be designed such that the system safety function can be accomplished with a single failure in the safety class system.

B. Types of Single Failures

Single failures may occur in either active or passive components. An active component is defined as one which requires physical motion of some sort to perform its intended function (e.g., motor operated valves or electrical relays). Failure of an active component means failure to provide the physical motion required for the safety function, not the failure of the passive parts (e.g., a valve body) of the active component. A passive component, such as a pipe, tank, accumulator, or electrical cable does not require physical motion to perform its function.

Active single failures may include failure to act or incorrect action when required, and spurious action or operation when not required. Examples of single active failures are failure of a valve to open or close in response to a valid signal and the operation of a motor-operated valve in the absence of a valid signal. Passive single failures may include ruptures or other failures of fluid system pressure boundaries (e.g., a pneumatic line), flow blockage in fluid systems, and electric short or open circuits

C. Application of the Single Failure Criterion

Reactor facilities are designed to accommodate a spectrum of postulated accidents without causing undue risk to health and safety. The basic procedure for conservatively determining the effects of a postulated accident and the effectiveness of safety class systems in mitigating potential consequences include the following steps:

1. The initiating event is specified. It may be assumed to occur in a safety class system (e.g., a reactivity transient caused by an uncontrolled positive reactivity insertion due to a fault in the reactivity control system).

2. Other failures that are a direct consequence of the initiating event are then assumed to take place.

3. In response to the postulated accident, safety class systems may be called into operation to mitigate the accident consequences. To ensure a conservative design of these systems, a single failure is assumed to occur in a safety class system, thereby diminishing its capabilities. Due to the inherent safety features of Category III reactors, few safety class systems are required.

4. Consequences of the postulated accident must then be demonstrated to be within the corresponding guidelines specified in Table 15.2-2.

10 CFR 50, Appendix A, states that fluid and electric systems are considered to be designed against an assumed single failure if neither a single failure of any active component (assuming passive components function properly) nor a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety function.

Some single failures, such as a failure of a standby electric power supply, may disable a protection system channel and any redundant safety system trains powered from the power supply. This multiple failure, resulting from a single occurrence, is still considered to be a "single failure."

The basic statement of the single failure criterion is concise; however, its effects are far reaching and often require interpretation to permit reasonable implementation in the design of reactor facility systems. Additional guidance on the application of the single failure criterion is found in USNRC Regulatory Guide 1.53, ANSI N658, and IEEE 379.

Safety class system failures may be classified as either detectable or undetectable failures. All potential single failures are detectable failures and may be discovered by periodic tests, anomalous indications, or alarms. For the purpose of system analysis, all potential undetectable failures should be identified by means of Failure-Mode-and-Effects-Analysis (FMEA). Undetectable failures differ from detectable failures in that they may exist in a safety class system for years. In fact, they could be built into the system by the manufacturer or installer. When undetectable failures are identified, the preferred course of action is to redesign the safety class system or the testing program to eliminate potential undetectable failures. An alternative approach is to perform the analysis of the effects of each single failure while assuming that all undetectable failures are in the failed mode. If this analysis indicates that the safety class system can still complete its required safety function and satisfactorily mitigate the consequences of the postulated design basis accident, then certain undetectable failures may be an acceptable feature of the system design.

For the purpose of applying the single failure criterion, the period following a postulated accident is generally divided into two phases - the "short term" and the "long term." During the short term, the facility should be designed to accommodate the postulated accident within the applicable consequence guidelines with the additional occurrence of: (1) any electrical single failure, or (2) any active fluid system failure (e.g., failure of a valve to close).

Passive fluid system single failure are not considered during the short term because the high integrity of reactor fluid systems make it very improbable that a passive fluid system component will experience random failure 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 an emergency core cooling 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, (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 residual heat removal system.

15.4 STANDARDS, CODES, AND GUIDES

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

A. Industry Standards and Codes

1. ANSI N658-1976, "Single Failure Criteria for PWR Fluid Systems"
2. ANSI N660 (ANS 51.4), Draft, "Criteria for Safety Related Operator Actions"
3. IEEE 379-1972 (ANSI N41.2), "Guide for the Application of the Single Failure Criterion of Nuclear Power Generating Station Protection Systems"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a LOCA for PWRs"
2. USNRC Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"
3. USNRC Regulatory Guide 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems"
4. USNRC Regulatory Guide 1.105, "Instrument Spans and Setpoints"

15.5 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. Pitkanen, P.H., "Hazards of Fast Burst Facilities," Nuclear Safety, Vol. 4, No. 1, September 1962

2. USNRC Standard Review Plan 2.2.3, "Evaluation of Potential Accidents," NUREG 75/087, September 1975
3. USNRC Standard Review Plan, 2.3.4, "Short Term Diffusion Estimates," *ibid.*
4. USNRC Standard Review Plan 15.2.6, "Loss of Non-Emergency AC Power to the Station Auxiliaries," *ibid.*
5. USNRC Standard Review Plan 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition," *ibid.*
6. USNRC Standard Review Plan 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," *ibid.*
7. USNRC Standard Review Plan 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)," *ibid.*
8. USNRC Standard Review Plan 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," *ibid.*
9. USNRC Standard Review Plan 15.7.2, "Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)," *ibid.*
10. USNRC Standard Review Plan 15.7.3, "Postulated Radioactive Releases Due to Liquid-Containing Tank Failures," *ibid.*
11. USNRC Standard Review Plan 15.7.4, "Radiological Consequences of Fuel Handling Accidents," *ibid.*
12. USNRC Standard Review Plan 15.7.5, "Spent Fuel Cask Drop Accidents," *ibid.*
13. USNRC Standard Review Plan 15.8, "Anticipated Transients Without Scram," *ibid.*
14. Branch Technical Position EICSB 18, "Application of the Single Failure Criteria to Manually Controlled Electrically Operated Valves," (in SRP, Appendix 7A), *ibid.*
15. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962
16. Slade, D.H., (Editor), "Meteorology and Atomic Energy - 1968," TID-24190, USAEC, 1968
17. Briggs, G.A., "Plume Rise," TID-25075, USAEC, 1969
18. Turner, D.B., "Workbook on Atmospheric Dispersion Estimates," AP-26, Revised, USEPA Office of Air Programs, 1970

19. 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

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

B. Pending Industry Standards and Codes

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

15.6 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

The following topical listing identifies areas in which the development of additional nuclear standards may provide a more consistent approach to Category III reactor facility design and modification:

A. Guidelines for a standard approach to calculating doses from postulated DOE reactor accidents. This guide should provide a standard means for defining source terms from a variety of fuel types used in Category III reactors. Additionally, guidance should be provided on a standard method for determining the fraction of the initial source term that is available for dispersion to other on-site and off-site locations.

CHAPTER 16

TECHNICAL SPECIFICATIONS

16.1 SCOPE

The ERDA Manual, Chapter 0540, "Safety of AEC-Owner Reactor," requires that the limitations for safe operation be specified in a Technical Specification document for each reactor, with the detailed requirements being commensurate with the potential risks that may arise from facility operation.

This chapter provides a listing of other guides and standards that are recommended for consideration when establishing technical specification limits for safe reactor operation.

16.2 APPLICABLE GENERAL CRITERIA

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

A. 10 CFR 50.36, "Technical Specifications"

B. 10 CFR 50, Appendix I, Section IV, "Guides on Technical Specifications for Limiting Conditions for Operation for Light Water Cooled Power Reactors Licensed Under 10 CFR 50"

16.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in establishing technical specifications for reactor facilities:

A. Industry Standards and Codes

1. ANSI N378-1974, "Standards for Development of Technical Specifications for Research Reactors"

B. RDT Standards

None

C. Guides

1. USNRC Regulatory Guide 1.16, "Reporting of Operating Information," Appendix A, Technical Specifications

2. USNRC Regulatory Guide 1.105, "Instrument Setpoints"

3. USNRC Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors"

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

16.4 SUPPLEMENTARY INFORMATION

The following documents provide information of a supplementary nature that may include useful background information for the designer. Industry codes and standards that are proposed for future development or 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," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and codes

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

16.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 17

QUALITY ASSURANCE

17.1 SCOPE

The ERDA Manual Chapter 0540, "Safety of AEC-Owned Reactors," requires that quality assurance programs be developed and implemented for reactor projects. The implementation of the Quality Assurance program is described in the ERDA Manual Chapter 0820, "Quality Assurance."

This chapter lists other standards and guides that are recommended for consideration when establishing QA programs for safety class components, systems and structures during the design, construction, and operation phases of a reactor facility. Non-safety class (non-nuclear) components, system and structures may utilize normal industrial quality assurance procedures in lieu of the codes, standards, and guides listed in this section, providing they meet the requirements of ERDAM 0820.

17.2 APPLICABLE 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, Appendix A, General Design Criterion 1, "Quality Standards and Records"

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

17.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in establishing a quality assurance program:

A. Industry Standards and Codes

1. General QA (applicable to all phases)

a. ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities"

b. ANSI N45.2.10-1973, "Quality Assurance Terms and Definitions"

c. ANSI N402-1976, "Quality Assurance Program for Research Reactors"

d. ANSI Z210.1-1973 (ASTM E380-72), "Metric Practice Guide"

- e. IEEE 268-1975, "Standard for Metric Practice"
- 2. Design Phase QA
 - a. ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"
- 3. Construction Phase QA
 - a. ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"
- 4. Operation Phase QA
 - a. ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants"
 - b. ANSI N399-1974, "Records and Reports for Research Reactors"

B. RDT Standards

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

- 1. General QA (applicable to all phases)
 - a. F1-2T, "Preparation of System Design Descriptions"
 - b. F2-2T, "Quality Assurance Program Requirements"
- 2. Design Phase QA
 - a. F2-9T, "Reliability Assurance"
- 3. Construction and Operation Phase QA
 - a. F2-4T, "Quality Verification Program Requirements"
 - b. F2-7T, "Acceptance Sampling Plans"
 - c. F3-2T, "Calibration Program Requirements"

C. Guides

- 1. General QA (applicable to all phases)
 - a. USNRC Regulatory Guide 1.74, "Quality Assurance Terms and Definitions"
 - b. USNRC Regulatory Guide 2.5, "Quality Assurance Program Requirements for Research Reactors"

2. Design Phase QA

a. USNRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)"

b. USNRC Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

3. Construction Phase QA

a. USNRC Regulatory Guide 1.28, "Quality Assurance program Requirements (Design and Construction)"

4. Operation Phase QA

a. USNRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)"

b. USNRC Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment"

17.4 SUPPLEMENTARY INFORMATION

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

A. Background Information

1. USNRC Standard Review Plan 17.1, "Quality Assurance During the Design and Construction Phase," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 17.2, "Quality Assurance During the Operations Phase"

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

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

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

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

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

8. IAEA Safety Guide SG-QA1, Draft, "Quality Assurance Program Preparation," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes

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

17.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None

CHAPTER 18

Distribution List

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