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MASTER

DESIGN GUIDE

FOR

CATEGORY IV REACTORS

LIQUID METAL REACTORS

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

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

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BNL 50831-IV

UC-80

(General Reactor Technology - TID-4500)

DESIGN GUIDE FOR CATEGORY IV REACTORS LIQUID METAL REACTORS

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July 1979

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***Science Applications, Inc., La Jolla, California**

**Prepared for the
DIVISION OF OPERATIONAL AND ENVIRONMENTAL SAFETY
UNITED STATES DEPARTMENT OF ENERGY**

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

Price: Printed Copy \$10.75; Microfiche \$3.00

July 1979

265 copies

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FOREWORD

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

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

DOE-owned reactors vary considerably in their design, siting, operating characteristics, and programmatic goals. A method has been developed for systematically and uniformly recommending codes, standards, guides, and general design criteria that apply to each facility in the diverse spectrum of DOE-owned reactors. The method is based on grouping the DOE-owned reactors into categories that are defined by generic reactor types. Within a given category, the DOE-owned reactors have generally similar design and/or operating characteristics. Table A-1 defines six generic categories of DOE reactors. Table A-2 lists the DOE reactors that are considered to be Category IV 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 facility systems. For each system, guidance is provided for determining if the system is required at a particular facility, and when the system is required, the Design Guide recommends major safety concepts and design philosophies to be implemented in the system in order to adequately protect health and safety.

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

Table A-1. ERDA Reactor Categories.

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

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

Table A-2. DOE Category IV Reactors

DOE Reactor	Maximum Steady-State Thermal Power Level (1)	Reactor Location
FFTF	400 MW	HEDL
EBR-II	62.5 MW	INEL

Note: (1) Based on TID-8200-R36, June 30, 1977,
"Nuclear Reactors Built, Being Built,
or Planned."

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 Category IV reactors. The Design Guide includes design guidelines which provide interpretations of the standards, codes, and guides in order to facilitate their applications to Category IV 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 guidance provided by the Design Guide, it remains for the designer to consult the appropriate referenced standards, codes, and guides in order to be fully appraised of all system design requirements. All of the applicable standards, codes, and guides that are recommended for consideration by the designer are available from the appropriate issuing organization.

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

Chapter 1

INTRODUCTION

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

1.1 PURPOSE

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

1.2 SCOPE

This Design Guide deals principally with the design and functional requirements of liquid metal cooled fast reactor (Category IV reactor) structures, components, and systems.

1.3 THE DESIGN GUIDE

1.3.1 Design Guide General Safety Philosophy

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

Because of their potentially large inventories of radioactive material, commercial power reactors provide multiple physical barriers to restrict the accidental release of radioactivity to the environment. These barriers generally include the fuel cladding, the reactor coolant system boundary, and a containment. Category IV DOE reactors are designed to provide the same set of physical barriers against the release of radioactivity to the environment. In addition, inherent safety features such as reduced potential source terms (due largely to lower operating power levels) and enhanced dispersion of released

radioactivity (due to remote siting) provide the required reasonable assurance that the DOE-owned reactor facility can be operated without undue risk to health and safety.

Table 1.3-1 provides a summary of the general safety philosophy for operation in terms of frequency of occurrence and radiological consequence guidelines corresponding to each facility operating condition (normal, upset, emergency and faulted). Additional consequence guidelines are included in Chapter 15. The facility operating conditions are discussed in detail in Section 3.9 and in Chapter 15.

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

A. Safety Class and Safety Function.

Reactor structures, components, and systems that are required for the protection of health and safety are those required for assuring safe shutdown and cooling of a reactor 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 for adequate protection of health and safety. Because of the complexity and varied designs of DOE-owned 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 Chapter 4 to 12 are those 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 meet quality standards commensurate with the importance of their safety function(s). A graded set of quality standards (see Table 3.1-1) establish the basis for defining the design requirements that apply to each of three nuclear safety classes (Safety Class 1, 2, and 3) and to a non-safety class (non-nuclear).

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

Table 1.3-1. Summary of the General Safety Philosophy for Facility Operation in Terms of Frequency of Occurrence and Radiological Consequences 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** ($\leq$ ERDAM 0524)	ALARA** ($\leq$ ERDAM 0524)
Upset	Incidents or occurrences of moderate frequency which may occur several times during the life of a facility	Loss of electric power, partial loss of core cooling, inadvertent reactor shutdown	ALARA** ($\leq$ ERDAM 0524)	ALARA** ($\leq$ ERDAM 0524)
Emergency	Infrequent incidents or occurrences which may occur during the lifetime of a facility	Loss of normal core cooling	Small fraction of 10CFR100	Small fraction of on-site faulted dose guidelines
Faulted	Events which are limiting faults and are not expected to occur during the life of a facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material and because they represent upper bounds on failures or accidents with a probability of occurrence sufficiently high to require consideration in design. Generally a postulated accident is not considered in design if the probability of its occurrence resulting in unacceptable consequences is less than 10^{-7} per year	Design basis reactivity accident	$\leq$ 10CFR100	$\leq$ 25 rem whole body, $\leq$ 300 rem thyroid, $\leq$ 150 rem bone, $\leq$ 75 rem lung, and $\leq$ 75 rem all other organs

** As low as reasonably achievable.

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. Additional guidelines are provided to assist the DOE contractor in determining the final safety class to which structures, components and systems at a specific reactor facility should be designed.

1. Minimum Safety Class.

The minimum safety classes recommended by this Design Guide are based on USNRC Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants." If the maximum conservatively calculated potential off-site doses due to a postulated failure of the system may exceed 0.5 rem to the whole body, the system, or appropriate portions of it, should be designed to Safety Class 1, 2, or 3. If the maximum conservatively calculated potential 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 to 12.

2. Final Safety Class.

Based on the accident analysis performed during the Safety Analysis Report (PSAR or FSAR) stage of facility review, the adequacy of the minimum safety class recommended by the Design Guide can be determined. Because of specific features of an individual Category IV reactor facility, analysis of some design basis accidents may indicate that the on-site and/or off-site radiological consequences are not within the guidelines in Table 1.3-1. Therefore, some system design changes may be necessary to ensure that a final safety class, equivalent to the recommended minimum safety class, will result in consequences within the guidelines of Table 1.3-1. Alternatively, it may be necessary to increase the quality and reliability of the system by designing it to a final safety class that is higher than the minimum safety class recommended by this 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 from the minimum safety class recommended by this Design Guide is required for either safety or programmatic reasons. Additional guidelines for determining final safety class are found in Chapter 15.

1.3.2 Use of the Design Guide

This guide has been written to provide standardized guidelines and recommendations pertinent to the design of new reactor facilities and to the modification of significant portions of existing facilities. For each system, guidance is provided for determining if the system is required at a particular facility and, if the system is required, the 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, codes, and USNRC Regulatory Guides, although written primarily for commercial power reactors, are also generally applicable to the design of Category IV reactors. Where necessary, design guidelines provide interpretations of the standards, codes, and guides in order to facilitate their application to Category IV 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 or modification. DOE Headquarters or the appropriate DOE Field Office may also approve the design or modification of a reactor facility to standards, codes, and guides other than those recommended by this Design Guide if the DOE contractor can demonstrate that the recommendations of this guide are overly restrictive for his particular application and/or that the alternative design guidelines adequately protect health and safety.

1.3.3 Design Guide Organization

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

Table 1.3-2 lists the systems that are typically associated with Category IV reactors and also references the Design Guide chapter or section that provides the detailed system design information. A comparable listing for Category IV reactors is also found in Table 3.2-2, which also summarizes the principal design codes, safety class, quality group, seismic category, and quality assurance requirements generally applicable to the design of typical Category IV reactor systems.

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

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

Figure 1.3-1. Design Guide Organization.

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

System	Design Guide Section
Reactor Core Assembly	4.1
Reactivity Control System	4.2
Primary Heat Transport System (including intermediate heat exchange loops)	5.1
Secondary Heat Transport System	5.2
Residual Heat Removal System	5.3
Reactor Building	6.2
Engineered Safety Feature Venti- lation System	6.3
Protection System	7.1
Non-Safety Instrumentation and Control System	7.2
Class IE Electric System	8.1
Non-Class IE Electric System	8.2
Cooling Water System	9.1
Fuel Handling and Storage System	9.2
Aux. Sodium & Purification Systems	9.3
Fire Protection System	9.4
Experimental Facilities	10
Radioactive Waste Systems	11
Radiation Protection System	12

(1) This systems listing should not be interpreted as a mandatory minimum systems requirement, but rather as a listing of systems that would be found at a hypothetical "average facility" and the corresponding Design Guide section or chapter references.

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

E. General Design Guidelines. This section provides basic guidance for the designer. Where existing codes, standards, and guides adequately address detailed design, functional, or procedural requirements, this section serves to highlight fundamental design criteria. Where necessary, interpretations are provided of the requirements of existing codes, standards, and guides in order to facilitate their application to Category IV reactors. If no applicable codes, standards, or guides have been developed, this section provides interim design guidance.

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. The majority of these standards, guides, and criteria were developed specifically for water-cooled reactors. The intent of these documents and requirements should be considered in the development of more specific criteria by the Category IV reactor staff.

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

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

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

1.3.4 Maintenance of the Design Guide

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

A program will be established for the periodic revision of the Design Guides and for the provision of additional information, as necessary, during the period between revisions. 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 guide, as necessary, at its next scheduled revision.

1.4 DEFINITIONS

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

A. ALARA.

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

B. Class IE.

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

C. Controlled Area.

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

D. Design Basis Accidents.

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

E. Health and Safety.

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

F. Off-Site Dose.

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

G. On-Site Dose.

Potential radiological exposure to persons within the exclusion area boundary.

H. Operating Basis Earthquake (OBE).

That earthquake which, considering the regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant. Features of the nuclear power plant necessary for continued operation without undue risk to health and safety are designed to remain functional during and following the vibratory ground motion produced by the OBE (10 CFR 100, Appendix A, Revised).

I. Operating Conditions.

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

J. 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 (1) the integrity of the primary coolant system boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 100 (10 CFR 100, Appendix A).

K. Safety Class.

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

L. Safety Function.

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

M. Safety Limit.

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

N. 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.

O. Severe Natural Phenomena.

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

P. Uncontrolled Area.

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

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CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Guidelines for Reactor Siting

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

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

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

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

2.1.2 Applicable 10 CFR Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the 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.

None.

B. RDT Standards.

None.

C. Guides.

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

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

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

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

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

2.1.4 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 2.1.1, "Site Location and Description," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 2.1.2, "Exclusion Area Authority and Control," *ibid.*

3. USNRC Standard Review Plan 2.1.3, "Population Distribution," *ibid.*

4. USNRC Standard Review Plan 2.2.1 and 2.2.2, "Location and Routes, Description," *ibid.*

5. USNRC Standard Review Plan 2.2.3, "Evaluation of Potential Accidents," *ibid.*

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

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

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

9. IAEA Safety Series 50-C-S, "Safety in Nuclear Power Plant Siting, A Code of Practice," International Atomic Energy Agency, Vienna

10. IAEA Safety Guide SG-S4, Draft, "Site Selection and Evaluation of Nuclear Power Plants with Respect to Population Distribution"

11. 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 Forecasting Future Population Distributions Surrounding Power Reactor Plants"

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

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

2.2.2 Applicable 10 CFR Siting Criteria

It is recommended that the following provisions of 10 CFR be considered in evaluating the suitability of the 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"

B. RDT Standards.

None.

C. Guides.

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

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

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

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

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

2.2.4 Supplementary Information

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

A. Background Information.

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

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

13. IAEA Safety Series 50-C-S, "Safety in Nuclear Power Plant Siting, A Code of Practice," International Atomic Energy Agency, Vienna

14. IAEA Safety Guide SG-S3a, Draft, "Meteorology-Climatology, Diffusion and Transport"

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

16. 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"

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

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

2.2.5 Topics for Future Standards Development

None.

2.3 HYDROLOGY

2.3.1 Guidelines for Reactor Siting

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

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

B. If groundwater is used as an essential source of water for safety-related equipment, the design basis for protection from natural and accident phenomena should consider the guidance for the ultimate heat sink (see Section 9.1).

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

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

2.3.2 Applicable 10 CFR Siting Criteria

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

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

B. 10 CFR 100, "Reactor Site Criteria"

2.3.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes.

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

2. ANSI/ANS 2.11-1978, "Guidelines for Evaluating Site-Related Geotechnical Parameters at Nuclear Power Sites"

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 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants"

5. USNRC Regulatory Guide 1.127, "Inspection of Water Control Structures Associated with Nuclear Power Plants"

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

2.3.4 Supplementary Information

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

A. Background Information.

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

1. USNRC Standard Review Plan 2.4.1, "Hydrologic Description," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 2.4.2, "Floods," *ibid.*

3. USNRC Standard Review Plan 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers," *ibid.*

4. USNRC Standard Review Plan 2.4.4, "Probable Dam Failures (Seismically Induced)," *ibid.*

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

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

7. USNRC Standard Review Plan 2.4.7, "Ice Effects," *ibid.*

8. USNRC Standard Review Plan 2.4.8, "Cooling Water Canals and Reservoirs," *ibid.*

9. USNRC Standard Review Plan 2.4.9, "Channel Diversions," *ibid.*

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

11. USNRC Standard Review Plan 1.4.11, "Low Water Considerations," *ibid.*

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

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

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

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

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

17. 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)

18. IAEA Safety Series 50-C-S, "Safety in Nuclear Power Plant Siting," International Atomic Energy Agency, Vienna

19. IAEA Safety Guide SG-S10A, Draft, "Design Basis Floods for Nuclear Power Plants on River Sites"

B. Pending Industry Standards and Codes.

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

2. ANSI N181 (ANS 2.9), Draft, "Standards for Evaluating Groundwater Supply for Nuclear Power Sites"

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

4. ANS 2.17, Draft, "Standards for Evaluating Radionuclide Transport in Groundwater for 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 Standard Development

None.

2.4 GEOLOGY AND SEISMOLOGY

2.4.1 Guidelines for Reactor Siting

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

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

B. Seismic features. Ground failure under dynamic loading, liquefaction, vibratory ground motion, site amplification, 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 CFR 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 2.11-1978, "Guidelines for Evaluating Site-Related Geotechnical Parameters at Nuclear Power Sites"

B. RDT Standards.

None.

C. Guides.

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

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

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

2.4.4 Supplementary Information

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

A. Background Information.

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

1. USNRC Standard Review Plan 2.5.1, "Basic Geologic and Seismic Information," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 2.5.2, "Vibratory Ground Motion," *ibid.*
3. USNRC Standard Review Plan 2.5.3, "Surface Faulting," *ibid.*
4. USNRC Standard Review Plan 2.5.4, "Stability of Subsurface Materials and Foundations," *ibid.*
5. USNRC Standard Review Plan 2.5.5, "Stability of Slopes," *ibid.*
6. NUREG 0406, "Method for Prediction of Strong Earthquake Ground Motion"
7. Cohee, G.V., "Tectonic Map of the United States," U.S. Geological Survey and American Association of Petroleum Geologists, 1962
8. Historical Earthquake Data File, National Geophysical and Solar Terrestrial Data Center, NOAA
9. "Earthquake History of the United States," Publication 41-1, U.S. Department of Commerce, NOAA
10. Lomenick, T.F., "Earthquakes and Nuclear Power Plant Design," ORNL-NSIC-28, Oak Ridge National Laboratory, July 1970
11. IAEA Safety Series 50-C-S, "Safety in Nuclear Power Plant Siting, A Code of Practice," International Atomic Energy Agency, Vienna
12. IAEA Safety Guide SG-S1, Draft, "Earthquakes and Associated Topics for Nuclear Power Plants"
13. IAEA Safety Guide SG-S2, Draft, "Seismic Analysis and Testing of Nuclear Power Plants," *ibid.*

B. Pending Industry Standards and Codes.

1. 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 IV 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 commercial power reactors, which may be designed to continue operating during some severe natural phenomena in order to supply electric power, Category IV 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., because of the scrambled condition of the reactor). Other safety class systems that are required for

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

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

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

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

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

Category IV 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 principal code requirements and design standards that are generally applicable to the design of Category IV reactor facility components, systems, and structures.

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 to prevent uncontrolled release of radioactivity with dose consequences in excess of the guidelines in Chapter 15 are designated Seismic Category I, and are specifically designed to withstand 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 displacement 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 Electric 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

Table 3.1-1. Principal Code and Design Requirements for Safety Class 1, 2, and 3 Components and for Non-Safety Class (Non-Nuclear) Components.

Component	Safety Class 1 (Quality Group A)	Safety Class 2 (Quality Group B)	Safety Class 3 (Quality Group C)	Non-Nuclear (Quality Group D)
Pressure vessels (1)	ASME B and PV Code, Section III, Division I, Class 1	ASME B and PV Code, Section III, Division I, Class 2	ASME B and PV Code, Section III, Division I, Class 3	ASME B and PV Code, Section VIII, Division I
Reactor containment pressure vessels (1)	--	ASME B and PV Code, Section III, Class MC	--	--
Pumps	ASME B and PV Code, Section III, Class 1	ASME B and PV Code, Section III, Class 2	ASME B and PV Code, Section III, Class 3	Manufacturer's Standards
Valves	ASME B and PV Code, Section III, Class 1	ASME B and PV Code, Section III, Class 2	ASME B and PV Code, Section III, Class 3	ANSI B16.5 and B31.1
Piping	ASME B and PV Code, Section III, Class 1	ASME B and PV Code, Section III, Class 2	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 2	ASME B and PV Code, Section III, Class 3	API-620 or equivalent
Atmospheric storage tanks	--	ASME B and PV Code, Section III, Class 2	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 1 and TEMA "R"	ASME B and PV Code, Section III, Class 2 and TEMA "R"	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"

(1) The ASME B and PV Code Section III, Division 2 is applicable to Safety Class 1, 2, and 3 concrete Vessels and containments.

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.

3.2.3 Determination of System Safety Class

Safety class systems are subdivided into safety classes or quality groups for the purpose of determining the applicable design and quality requirements. The terms Safety Class and Quality Group are interchangeable with Safety Classes 1, 2, and 3 corresponding directly to Quality Groups A, B, and C. Non-Safety Class (or non-nuclear) components are in Quality Group D. USNRC Regulatory Guide 1.26 requires that a system be designed to a safety class if the conservatively calculated potential off-site doses due to failure of the system can exceed 0.5 rem to the whole body. If these doses 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 IV reactor facility components, systems, and structures.

Applying the general guidelines of Table 3.2-1, Table 3.2-2 lists the likely minimum safety classes of reactor systems that are typical of Category IV reactors. Table 3.2-2 is not to be interpreted as a listing of required systems for Category IV reactors. It is rather a listing of systems that would be found at a hypothetical "typical facility." Individual facilities may vary substantially from this listing while still adequately protecting health and safety. Because of differences between DOE Category IV reactors and commercial Liquid Metal Fast Breeder Reactor (LMFBR) designs, the Table 3.2-1 and 3.2-2 listings do not necessarily correspond to the safety class guidelines that may be applicable to commercial LMFBRs. Category IV reactors are, however, designed to provide a defense-in-depth, comparable to that required for a commercial LMFBR design, against the occurrence of a serious accident.

Safety class changes within a system or between interconnecting systems are permitted subject to certain restrictions. In general, changes in safety class may occur at pressure boundaries (e.g., heat exchangers), or positive isolation devices (e.g., heat exchangers), or positive isolation devices (e.g., normally closed valves or valves capable of automatic closure). Because of the large number of potential system configurations, the safety class changes should be reviewed individually.

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.

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

SAFETY CLASS 1 (Quality Group A)

SC-1 applies to the following components:

- A. Those which comprise part of the reactor coolant boundary (RCB).
The RCB includes components such as the vessel, piping, pumps, tanks and valves that are:
 - 1. Part of the Reactor Coolant System, or
 - 2. Connected to the Reactor Coolant System, up to and including any and all of the following:
 - a. The second of two valves normally closed or automatically isolable during normal reactor operation,
 - b. The passive barrier between the reactor coolant and the intermediate coolant.
- B. Those which perform scram functions under any plant conditions.
- C. Those which maintain core geometry or provide core support and whose failure could initiate an accident condition.

SAFETY CLASS 2 (Quality Group B)

SC-2 applies to the following components:

- A. Those which are a part or extension of the reactor containment boundary
- B. Those which are required to maintain an adequate reactor coolant inventory following a reactor coolant boundary leak (e.g., guard vessel)
- C. Those whose single failure would cause a loss of safety function of other SC-2 components.

SAFETY CLASS 3 (Quality Group C)

SC-3 applies to the following components:

- A. Those which are required to remove residual heat from the reactor or from spent fuel
- B. Those which are extensions of the reactor coolant boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.
- C. Those whose failure could result in loss of safety function of another component (e.g., cooling water systems).
- C. Those components, systems or structures that are not SC-1 or SC-2, but contain or may contain radioactive material and have the potential for causing off-site doses in excess of 0.5 rem to the whole body or its equivalent to any part of the body, but less than 10CFR100 dose guidelines.

- Note (1) These guidelines for determining the minimum safety class are based on USNRC Regulatory Guide 1.26. General Guidance for determining the final safety class are presented in Chapter 15.
- (2) Components, system 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.

Table 3.2-2 Typical Equipment Classification for Category IV Reactors

System	Design Guide Section	Safety Class	Quality Group	Seismic Category	10CFR50 Appendix B	Principal Design Code or Standard
Reactor Core Assembly	4.1	--	-	I	Applies	-
Reactivity Control System	4.2	-	-	I	Applies	-
Primary Heat Transport System	5.1	(a)	(a)	I	Applies	ASME B&PV Code, Section III
Secondary Heat Transport System	5.2	NN ^(a)	D	-	-	ASME B&PV Code, Section III
Residual Heat Removal System	5.3	3	C	I	Applies	ASME B&PV Code, Section III
Reactor Building	6.2	2	B	I	Applies	ASME B&PV Code, Section III
ESF Ventilation System	6.3	3	C	I	Applies	ASME B&PV Code, Section III
Protection System	7.1	-	-	I	Applies	IEEE 279
Non-Safety I&C System	7.2	-	-	-	-	NEMA ICS-1970
IE Electric Power System	8.1	-	-	I	Applies	IEEE 308
Non-IE Electric Power System	8.2	-	-	-	-	ANSI C1-1975, NEMA MG 1-1972
Cooling Water System	9.1	3	C	I	Applies	ASME B&PV Code, Section III
Fuel Handling & Storage System	9.2	3	C	I	Applies	ASME B&PV Code, Section III
Auxiliary Sodium and Purification Systems	9.3	3	C	I	Applies	ASME B&PV Code, Section III ANSI B31.1
Fire Protection System	9.4	3	C	I	Applies	NFPA 802-1974
Experimental Facilities	10	(b)	(b)	(b)	(b)	-
Radioactive Waste System	11	3	C	I	Applies	ASME B&PV Code, Section III
Radiation Protection System	12	3	C	I	Applies	-

Notes: (a) See guidelines, Table 5.1-1.

(b) Experimental facilities may interface with many facility systems, including the reactor core assembly, reactor coolant system boundary, instrumentation, control and electrical 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.

Determining the safety class of structures, components, and systems provides the designer with the information necessary for defining the portions of the ASME Boiler and Pressure Vessel Code, or other applicable guidelines, that should be considered in the design of fluid systems and mechanical and structural components. The methods for determining the design and quality standards applicable to ASME code and non-ASME code equipment are described in the following paragraphs:

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

Structures, components, and systems designed to the ASME Boiler and Pressure Vessel Code may be Safety Class 1, 2, or 3 or they may be non-nuclear. As illustrated in Table 3.1-1, the safety class of the fluid system determines the portion of the ASME code that should be considered in the design of the system. Safety Class 1, 2, and 3 systems designed to the ASME code are subject to the quality assurance requirements of 10 CFR 50, Appendix B and ANSI N45.2. Non-nuclear systems designed to the ASME code are subject to normal industrial QA practices.

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 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 to the environment.

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

Safety class systems, components, and structures that are not subject to the ASME code are subject to the quality assurance requirements of 10CFR 50, Appendix B and ANSI N45.2. Non-nuclear systems are subject to normal industrial QA practices.

3.2.4 Applicable 10 CFR Criteria

The following General Design Criteria and other provisions of 10 CFR were developed specifically for water-cooled reactors. The intent of these requirements should be considered in the development of more specific GDC by the

Category IV reactor project staff 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/ANS 4.1-1978, "Design Basis Criteria for Safety Systems 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 Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants"

3.2.6 Supplementary Information

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

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

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

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

B. Pending Industry Standards and Codes.

1. ANS 54.6, Draft, "LMFBR Safety Classification and Related Requirements"

2. ANS 54.11, Proposed, "Application of Risk Limit Guidelines for LMFBR Design"

3.2.7 Topics for Future Standards Development

None.

3.3 WIND AND TORNADO LOADINGS

For the safety class systems identified in Section 3.2, the following wind loading design guidelines should be considered:

3.3.1 Design Guidelines

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.

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 are likely to exceed any other wind loadings. 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

The following 10 CFR 50, Appendix A, General Design Criterion was developed specifically for water-cooled reactors. The intent of these requirements should be considered in the development of more specific GDC by the reactor staff for designing the 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 2.12-1978, "Guidelines for Combining Natural and External Man-made Hazards at Power Reactor Sites"

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

B. RDT Standards.

None.

C. Guides.

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

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

3.3.4 Supplementary Information

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

A. Background Information.

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

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

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

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

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

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

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

B. Pending Industry Codes and Standards.

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

3.3.5 Topics for Future Standards Development

None.

3.4 WATER LEVEL (FLOOD) DESIGN

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

3.4.1 Flood Design

Components, systems, and structures should be designed to withstand the effects of flooding if they must function during and after floods, including flooding induced by failure of water systems, 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.

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

The following 10 CFR 50, Appendix A, General Design Criterion was developed specifically for water-cooled reactors. The intent of these requirements should be considered in the development of more specific GDC by the reactor staff for designing a facility to withstand the effects of floods.

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

3.4.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in designing a 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 2.12-1978, "Guidelines for Combining Natural and External Man-made Hazards at Power Reactor Sites"
3. ANSI/ANS 58.3-1977 (N182), "Physical Protection for Systems and Components Important to Safety"

- B. RDT Standards.

None.

- C. Guides.

1. USNRC Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"
2. USNRC Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants"
3. USNRC Regulatory Guide 1.135, "Normal Water Level and Discharge at Nuclear Power Plants"

3.4.4 Supplementary Information

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

- A. Background Information.

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

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

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

B. Pending Industry Standards and Codes.

1. ANSI N515 (ANS 2.4), 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 Missiles Generated at the Reactor Facility

Among the facility-generated missiles to be considered as potential hazards are those arising from failure of experimental facilities, high-speed rotating machinery, or high energy fluid systems components. Examples of potential missiles are steam system valve bonnets, bolts, instrument wells, gas cylinders, and fan blades.

Equipment with a potential for generating missiles should be located such that safety class systems will not be endangered. Where the physical separation of systems does not provide adequate missile protection, barriers or other protective structures should be provided to ensure that the safety class systems can withstand the effects of potential missiles. Appropriate credit may be taken for the protection afforded by a biological shield surrounding the major reactor coolant system components. The biological shield may effectively limit the potential effects of some facility-generated missiles.

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. Safety class systems of a facility should be designed to

withstand the effects of missiles or should be protected by structures or missile barriers. In particular, the design of the reactor building and the spent fuel storage area should prevent potential externally generated missiles from contacting safety class systems and spent fuel assemblies respectively.

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

3.5.3 Other Externally Generated Missiles

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

3.5.4 Applicable General Design Criteria

The following 10 CFR 50, Appendix A, General Design Criteria were developed specifically for water-cooled reactors. The intent of these requirements should be considered in the development of more specific GDC by the reactor staff in designing a facility to withstand the effects of missiles.

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

3.5.5 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 2.12-1978, "Guidelines for Combining Natural and External Man-made Hazards at Power Reactor Sites"
 - 3. 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 1.27, "Ultimate Heat Sink"
3. USNRC Regulatory Guide 1.91, Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites"
4. USNRC Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles"

3.5.6 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 3.5.1.1, "Internally Generated Missiles (Outside Containment)," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 3.5.1.2, "Internally Generated Missiles (Inside Containment)," *ibid.*
3. USNRC Standard Review Plan 3.5.1.3, "Turbine Missiles," *ibid.*
4. USNRC Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," *ibid.*
5. USNRC Standard Review Plan 3.5.1.5, "Site Proximity Missiles (Except Aircraft)," *ibid.*
6. USNRC Standard Review Plan 3.5.1.6, "Aircraft Hazards," *ibid.*
7. USNRC Standard Review Plan 3.5.2, "Structures, Systems, and Components to be Protected from Externally Generated Missiles," *ibid.*
8. USNRC Standard Review Plan 3.5.3, "Barrier Design Procedures," *ibid.*
9. Branch Technical Position AAB 3-2, "Tornado Design Classification (Appended to SRP 3.5.1.4)," *ibid.*

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

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

12. Gwaltney, R.C., "Missile Generation and Protection in Light Water Reactor Plants," ORNL-NSIC-22, Oak Ridge National Laboratory, September 1969

13. IAEA Safety Guide SG-D4, Draft, "Safety Guide on Protection Against Internally Generated Missiles and Their Secondary Effects in Nuclear Power Plants," International Atomic Energy Agency, Vienna

14. IAEA Safety Guide SG-D5, Draft, "Man-Induced Events"

B. Pending Industry Standards and Codes.

None.

3.5.7 Topics for Future Standards Development

None.

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

For safety class systems identified in Section 3.2, the following guidelines for protection against the dynamic effects of pipe rupture should be considered:

3.6.1 Guidelines for Protection Against Pipe Rupture

Structures, systems, and components required for 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 protected against dynamic effects that may result from pipe rupture.

In general, there is no significant pipe whip hazard in the primary heat transport system (primary and intermediate sodium heat transport loops). Category IV reactors generally operate with coolant temperatures hundreds of degrees below the boiling point. Therefore, the liquid metal portions of the cooling loops need not be pressurized. Generally, Category IV reactors operate with reactor coolant system pressure near ambient pressure. The pressure exerted on the coolant system boundary is only that of the pump head required for forced coolant circulation plus the static head of liquid metal coolant. For these reasons the liquid metal portions of the coolant system have little stored thermodynamic energy. Small leaks, should they occur, have little likelihood of propagation into larger ones. Moreover, the low stored energy in the primary

heat transport system does not, of itself, generate pressure within the containment structure in case of leakage (sodium chemical reactions and fires do represent a source of a post-leakage containment pressure transient).

A steam generating secondary heat transport system (if provided) is a high energy fluid system and, as such, represents a potential pipe whip hazard. This system would generally be located outside the containment and, as such, should not represent a direct pipe whip hazard to equipment within the containment.

In general, breaks are postulated to occur at terminal ends of piping runs, and at appropriate intermediate locations selected on the basis of stress levels, usage factors, and other reasonable factors.

Methods of protection against the effects of pipe rupture include physical separation of potential sources of pipe rupture from safety class systems (e.g., by placing entire secondary heat transport system outside containment), installation of pipe whip restraints, and installation of barriers or other protective structures around safety class systems.

The ASME Boiler and Pressure Vessel Code, Section III, addresses the methods for assuring the pressure-retaining integrity in fluid piping systems. Following a fluid system failure, any dynamic effects (e.g., plastic hinge formation and pipe whipping) of that part of the piping system remaining intact may create a significant hazard.

In determining the need to provide protection against the mechanical effects of pipe whip on safety class systems, it should be considered that an unrestrained whipping pipe may be capable of rupturing impacted pipes of smaller nominal pipe size and causing through-wall cracks in larger nominal pipe sizes with thinner wall thicknesses (see USNRC Branch Technical position APCS 3-1). In addition to the mechanical effects of pipe whip, the effects of jet impingements and the hostile accident environment (fluid sprays, flooding, high temperature, and compartment pressurization) on safety class systems and any protective structures should be considered. Requirements for system design to withstand postulated accident environments are found in Section 3.11.

3.6.2 Applicable General Design Criterion

The following 10 CFR 50, Appendix A, General Design Criterion was developed specifically for water-cooled reactors. The intent of these requirements should be considered by the reactor staff in the development of more specific GDC for designing a facility to withstand the effects of pipe whip.

A. GDC 4, "Environment and Missile Design Basis"

3.6.3 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 pipe whip:

A. Industry Standards and Guides.

1. ANSI N176 (ANS 58.2), Draft, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture"

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

B. RDT Standards.

None.

C. Guides.

None.

3.6.4 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," *ibid.*

3. Branch Technical Position APCS 3-1, "Protection Against Postulated Piping Failures in Piping Systems Outside Containment," (Appended to SRP 3.6.1), *ibid.*

4. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," (Appended to SRP 3.6.2), *ibid.*

B. Pending Industry Standards and Codes.

1. ANS 58.2, Draft, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture"

3.6.5 Topics for Future Standards Development

None.

3.7 SEISMIC DESIGN

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

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 correspond to maximum vertical and horizontal ground acceleration of 1.0 g and an accompanying displacement of 36 inches, without considering soil-structure interaction.

3.7.1.2 Critical Damping Values

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

3.7.2 Seismic System Analysis

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

A. Dynamic analysis of systems unless the equivalent static load method can be justified based on simple system configuration and conservative analytical results.

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

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

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

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

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

3.7.3 Seismic Instrumentation

Seismic instrumentation should be provided to permit the determination of the seismic response of safety class systems during an earthquake and to permit a comparison of such response with the design basis response. Such a comparison may be needed to decide whether the facility can safely continue operation following an earthquake. If the design basis response is exceeded, the reactor would 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

The following 10 CFR 50, Appendix A, General Design Criterion and provisions of 10 CFR 100 should 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/ANS 2.2-1978, "Earthquake Instrumentation Criteria for Nuclear Power Plants"

2. ANSI/ANS 2.10-1979, "Guidelines for Retrieval, Review, Processing, and Evaluation of Records Obtained from Seismic Instrumentation"

B. RDT Standards.

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

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

C. Guides.

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

3.7.6 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 3.7.1, "Seismic Input," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 3.7.2, "Seismic System Analysis," *ibid.*
3. USNRC Standard Review Plan 3.7.3, "Seismic Subsystem Analysis," *ibid.*
4. USNRC Standard Review Plan 3.7.4, "Seismic Instrumentation," *ibid.*
5. NUREG-0406, "Methods for Prediction of Strong Earthquake Ground Motion"
6. WASH 1361, "Safety-Related Site Parameters for Nuclear Power Plants," January 1975
7. Werner, S.D., "Study of Earthquake Input Motions for Seismic Design," TID-25438, June 1970

8. 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

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

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

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

B. Pending Industry Standards and Codes.

None.

3.7.7 Topics for Future Standards Development

None.

3.8 DESIGN OF SEISMIC CATEGORY I AND NON-SEISMIC CATEGORY I STRUCTURES

Section 3.2 provides guidelines for determining if a structure should be designed to Seismic Category I standards. The following design guidelines are applicable to reactor facility structures:

3.8.1 Seismic Category I Concrete or Steel Reactor Building

3.8.1.1 Design Recommendations

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 the design of the reactor building are presented in Section 6.2 along with codes and standards that are specifically applicable to containments.

Loading combinations and the design and analytical procedures to be considered in the design of Seismic Category I structures are discussed in several standards and codes listed in the following subsections. The loads normally applicable to 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, including dead loads, live loads, thermal loads, and hydrostatic loads.

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 and accident plant conditions. The Design Basis Accidents listed in Chapter 15 should be considered in reactor building design, including accidents involving various fluid system pipe ruptures (e.g., sodium line break in an intermediate heat transport loop). Loads induced on the reactor building by such accidents include elevated temperatures and pressures and possible localized loads due to missile impact. Other accidents may be more limiting for other Seismic Category I structures.

3.8.1.2 Testing and Inservice Surveillance Requirements

The acceptance testing of concrete containment structures should include a pressure test which increases internal containment pressure to at least 1.15 times the containment design pressure. Details for such a test are described in USNRC Regulatory Guide 1.57. A containment should be designed to facilitate periodic testing and inspection throughout the life of the facility in order to verify that leak rates, filter efficiencies, and other safety-related parameters are within limits specified for the facility.

3.8.2 Concrete and Steel Internal Structures of a Seismic Category I Steel or Concrete Reactor Building

3.8.2.1 Design Requirements

Internal structures may have safety functions for which their structural integrity is important. For example, by providing support during normal operations and during seismic disturbances, they should prevent the occurrence of a pipe break in the intermediate heat transport loop. If such an accident does occur, however, the internal structures should act to mitigate the consequences of the pipe break by protecting the nearby safety-related components from effects of the accident (e.g., sodium fires or immersion in liquid sodium).

The geometry and complexity of reactor building internal structures may vary significantly among reactor facilities; however, the following is a list of typical concrete or steel internal structures:

- A. Reactor supports.
- B. Reactor biological shield.
- C. Other shield walls forming compartments for heat exchangers.
- D. Supports for intermediate heat exchange loop components.

E. Supports for overhead crane.

Loading combinations and design and analytical procedures to be considered in the design of these structures are discussed in Section 3.9. Loads normally applicable to reactor building internal structures are similar to those considered in Subsection 3.8.1.1 for the reactor building.

3.8.2.2 Testing and Inservice Surveillance Requirements

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

3.8.3.1 Design Recommendations

Structures other than a Category I reactor building and its internal structures may have safety functions associated with protection of safety class components or systems such as emergency electric generators, essential heat exchangers, or water supplies. Design requirements for these structures are specified in ANSI A89.1-1972 and in AISC steel specifications. Loads normally applicable to these seismic Category I structures are similar to those considered in Subsection 3.8.1.1 for the reactor building.

3.8.3.2 Testing and Inservice Surveillance Requirements

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

3.8.4 Non-Seismic Category I Structures

Structures that are not required to be seismic Category I may be designed to normal industrial standards for buildings. These structures are not designed to withstand the OBE or SSE and are not designed to protect enclosed components and systems from the effects of severe natural phenomena. Non-seismic Category I structures do not require periodic testing or in-service surveillance.

3.8.5 Applicable General Design Criteria

The following 10 CFR 50, Appendix A, General Design Criteria were developed specifically for water-cooled reactors. The intent of these requirements should be considered by the reactor staff in the development of more specific GDC for the design of seismic Category I buildings. 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"

3.8.6 Standards Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the design of reactor facility 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. Seismic Category I Structures.

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. ANSI/ANS 2.12-1978, "Guidelines for Combining Natural and Man-Made Hazards at Power Reactor Sites"

f. ACI 349-76, "Code Requirements for Nuclear Safety-Related Concrete Structures," American Concrete Institute

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

h. ASME Boiler and Pressure Vessel Code, Section III, Division 1 and Division 2 General Requirements

i. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, "Class MC Components"

j. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, "Component Supports"

k. ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Concrete Reactor Vessels and Containments"

l. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"

2. Non-Seismic Category I Structures.

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. Seismic Category I Structures.

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

2. Non-Seismic Category I Structures.

None.

C. Guides.

1. Seismic Category I Structures.

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

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

c. USNRC Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments"

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

e. USNRC Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components"

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

g. USNRC Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment Components"

h. USNRC Regulatory Guide 1.103, "Post-tensioned Prestressing Systems for Concrete Reactor Vessels and Containments"

i. USNRC Regulatory Guide 1.107, "Qualification for Cement Grouting for Prestressed Tendons in Containment Structures"

j. USNRC Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports"

k. USNRC Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate- and Shell-Type Component Supports"

l. USNRC Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)"

2. Non-Seismic Category I Structures.

None.

3.8.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 3.8.1, "Concrete Containment," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 3.8.2, "Steel Containment," *ibid.*

3. USNRC Standard Review Plan 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," *ibid.*

4. USNRC Standard Review Plan 3.8.4, "Other Seismic Category I Structures," *ibid.*

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

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

B. Pending Industry Standards and Codes.

None.

3.8.8 Topics for Future Standards Development

None.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

The spectrum of postulated design basis events define four facility operating conditions for which corresponding stress levels are defined. 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 consequence 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 and pressure 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 ASME Code Class Components, Component Supports, and Core Support Structures

Safety Class 1, 2, and 3 components designed to the requirements of the ASME code (including pressure vessels, piping, valves, and pumps) are required to operate during and after a spectrum of design basis events. These events range from normal occurrences, such as refueling and startups, to severe abnormal events such as pipe breaks and reactivity accidents. The ASME Boiler and Pressure Vessel Code specifies four service loading 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 specifies the service level A, B, C, and D load limits for ASME code class components. The basis for these limits is summarized in Table 3.9-3.

For reactor facilities, the loading combinations to be considered in determining the proximity to the appropriate service load limits are based on USNRC Regulatory Guide 1.48. These loading combinations are summarized in Table 3.9-4. The Operating Basis Earthquake (OBE) is generally considered to be one-half the magnitude of the Safe Shutdown Earthquake (SSE). The OBE is superimposed on the loads resulting from upset events in order to determine the total upset load and the margin to the ASME Boiler and Pressure Vessel Code load limits for level B service. Likewise, the SSE is superimposed on the loads resulting from faulted events (e.g., a design basis reactivity accident) in order to determine the total faulted load and the margin to the load limits for level D service. 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

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, refueling
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
Emergency	Infrequent incidents or occurrences which may occur during the lifetime of a facility	Loss of normal core cooling
Faulted	Events which are limiting faults and are not expected to occur during the life of a facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material and because they represent upper bounds on failures or accidents with a probability of occurrence sufficiently high to require a consideration in design. Generally a postulated accident is not considered in design if the probability of its occurrence resulting in unacceptable consequences is less than 10^{-7} per year	Design Basis Reactivity Accident

Table 3.9-2. Design Limits for ASME Code Class Components
(Based on USNRC Regulatory Guide 1.48)

USNRC Regulatory Guide 1.48 Paragraph	Affected Components	Operating Condition (a,b)	ASME Service Loading Level (c)	ASME Boiler and Pressure Vessel Code Section III, Division 1 Paragraph References for Stress and Pressure Limits	General Notes
C.1	ASME Code Class 1 vessels and piping	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	Vessels: NB-3222 Piping: NB-3653 NB-3223 NB-3654 NB-3224 NB-3655 NB-3225 NB-3656	
C.2	ASME Code Class 1 pumps and valves (non- active) designed by analysis	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	NB-3222 NB-3223 NB-3224 NB-3225	
C.3	ASME Code Class 1 valves (non-active) de- signed by standard or alternative design rules	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	NB-3525 NB-3525 NB-3526 NB-3527	
C.4	ASME Code Class 1 pumps and valves (active) designed by analysis	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	NB-3222 NB-3223 NB-3224 NB-3225	(d)
C.5	ASME Code Class 1 valves (active) designed by standard or alterna- tive design rules	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	NB-3525 NB-3525 NB-3526 NB-3527	(d)
C.6	ASME Code Class 2 and 3 vessels	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	Class 2: NC-3321 Class 3: ND-3321	

Table 3.9-2. Design Limits for ASME Code Class Components
(Based on USNRC Regulatory Guide 1.48)
(Continued)

USNRC Regulatory Guide 1.48 Paragraph	Affect Components	Operating Condition ^(a,b)	ASME Service Loading Level ^(c)	ASME Boiler and Pressure Vessel Code Section III, Division 1 Paragraph References for Stress and Pressure Limits	General Notes
C.7	ASME Code Class 2 vessels designed to alternate design rules for vessels	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	NC-3217	
C.8	ASME Code Class 2 and 3 piping	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	Class 2: NC-3611-2, Class 3: ND-3611-2	
C.9	ASME Code Class 2 and 3 pumps (non-active)	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	Class 2: NC-3423, Class 3: ND-3423	
C.10	ASME Class 2 and 3 pumps (active)	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	Class 2: NC-3423, Class 3: ND-3423	(d)
C.11	ASME Code Class 2 and 3 valves (non-active)	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	Class 2: NC-3512, Class 3: ND-3512 NC-3520 ND-3520 NC-3520 ND-3520 NC-3520 ND-3520	
C.12	ASME Code Class 2 and 3 valves (active)	Normal Upset + $\frac{1}{2}$ SSE Emergency Faulted + SSE	A B C D	Class 2: NC-3512, Class 3: ND-3512 NC-3520 ND-3520 NC-3520 ND-3520 NC-3520 NC-3520	(d)

3-25(c)

Table 3.9-2. Design Limits for ASME Code Class Components
(Based on USNRC Regulatory Guide 1.48)
(Continued)

NOTES:

- (a) SSE = Safe shutdown earthquake; OBE = operating basis earthquake = $\frac{1}{2}$ SSE.
- (b) Each facility should determine the specific transients or events to be considered under each plant operating condition.
- (c) Service loading levels are defined in the ASME Boiler and Pressure Vessel Code Section III, Division 1.
- (d) These stress limits for active components do not assure the operability of the component. Assurance of operability under all design loading combinations should be provided per footnote 6 of USNRC Regulatory Guide 1.48.

Table 3.9-3. Basis for ASME Boiler and Pressure Vessel
Code Section III Service Loading Limits^(a)

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:

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

Table 3.9-4. Typical Design Loading Combinations for ASME Code Class Components and Supports

Operating Condition	Load Combination ^(a)	
	Equation ^(b)	Remarks
Normal	$P + DW + TNL$	Normal operating transient temperature, pressure, and piping loads to be considered
Upset	$P + DW + TNL$	Upset operating condition transients to be considered
	$P + DW + TNL + \frac{1}{2}SSE$	OBE seismic condition, including transient pressure, temperature, and piping loads resulting from the OBE, to be considered
Emergency	$P + DW + TNL$	Emergency transient temperature, pressure, and piping loads to be considered.
Faulted	$P + DW + TNL$	Faulted transient temperature, pressure, and piping loads to be considered
	$P + DW + TNL + SSE$	Faulted Design Basis Accidents and SSE to be combined concurrently; transient temperatures and pressures and piping loads to be considered

(a) The combination of seismic loads with upset and faulted loads is based on the recommendations of USNRC Regulatory Guide 1.48.

(b) The terms used in the equations are defined as follows:

P = Pressure loading for specified operating condition, including the effects of transients.

DW = Loads due to dead weight (piping loads, equipment loads, etc.).

TNL = Loads due to thermal expansion of piping attached to components or supports, including transient effects for specified operating condition.

OBE = Loads due to Operating Basis Earthquake (OBE) = $\frac{1}{2}$ SSE.

SSE = Loads due to Safe Shutdown Earthquake (SSE)

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 Safety Class Mechanical Components Not Covered by the ASME Code

Safety Class 1, 2, and 3 mechanical components not covered by the ASME code may include the following:

- A. Fuel elements.
- B. Mechanical portions of the reactivity control system.
- C. Portions of reactor internal structures.
- D. Fuel handling equipment.
- E. Thermal insulation systems.

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

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

3.9.3 Operability Tests for Safety Class Mechanical Components

The operability of all ASME Code class and non-ASME Code class mechanical components that must perform a mechanical motion in the performance of a safety function should be assured. These Safety Class 1, 2, and 3 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 analysis or by a combination of these methods.

3.9.4 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 and 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 listed in Subsection 3.9.6 and in Chapters 4 to 12, where appropriate.

3.9.5 Applicable 10 CFR Criteria

In addition to the General Design Criteria listed in the chapters describing the individual systems, the following provisions of 10 CFR 50 were developed specifically for water-cooled reactors. The reactor staff should consider the intent of these requirements in developing more specific GDC for the design of mechanical systems and components.

A. GDC 1, "Quality Standards and Records"

B. 10 CFR 50.5a, "Codes and Standards"

3.9.6 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. Safety Class Mechanical Components.

a. ASME Boiler and Pressure Vessel Code, Section III, Division 1 and Division 2, "General Requirements"

b. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, "Class 1 Components"

c. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, "Class 2 Components"

d. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection ND, "Class 3 Components"

e. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF, "Component Supports"

f. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, "Core Support Structure"

g. ANSI/ANS 4.1-1978, "Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations"

2. Non-Safety Class Mechanical Components.

a. ASME Boiler and Pressure Vessel Code, Section VIII, "Pressure Vessels"

b. ANSI B31.1-1977, "Power Piping"

c. Other ANSI B-Series Standards for Mechanical Components

B. RDT Standards.

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

1. Safety Class Mechanical Components.

a. E15-2NB-T, "Class 1 Nuclear Components" (Supplements ASME Code, Section III, Subsections NA and NB)

b. E15-2NC-T, "Class 2 Nuclear Components" (Supplements ASME Code, Section III, Subsections NA and NC)

c. E15-2ND-T, "Class 3 Nuclear Components" (Supplements ASME Code, Section III, Subsections NA and ND)

d. E15-2NE-T, "Class MC Nuclear Components" (Supplements ASME Code, Section III, Subsections NA and NE)

e. Other RDT E-series standards for specifications for various components for liquid metal reactor systems

f. F6-11T, "Fabrication and Installation of Piping Subassemblies for Liquid Metal Service"

g. F9-4T, "Requirements for Construction of Nuclear System Components at Elevated Temperatures" (Supplement to ASME Code Cases 1592, 1593, 1594, 1595 and 1596)

h. F9-5T, "Guidelines and Procedures for Design of Nuclear System Components at Elevated Temperature"

2. Non-Safety Class Mechanical Components.

None.

C. Guides.

1. Safety Class Mechanical Components.

a. USNRC Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components"

b. USNRC Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III, Division 1"

c. USNRC Regulatory Guide 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports"

d. USNRC Regulatory Guide 1.130, "Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports"

2. Non-Safety Class Mechanical Components.

None.

3.9.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Mechanical Systems and Components," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," ibid.

3. USNRC Standard Review Plan 3.9.6, "Inservice Testing of Pumps and Valves, ibid.

B. Pending Industry Standards and Codes.

1. ANS 54.11, Proposed, "Application of Risk Limit Guidelines for LMFBR Design"

2. ANS 54.12, Proposed, "Event Categorization Guidelines for LMFBR Design"

3.9.8 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 electric 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.

Supports for Seismic Category I instrumentation and electrical equipment should be tested with the equipment in place or with simulated equipment. The combined stresses on the component supports should be within the limits allowed by the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF.

3.10.2 Applicable General Design Criteria

The following 10 CFR 50, Appendix A, General Design Criterion was developed specifically for water-cooled reactors. The intent of these requirements should be considered by the reactor staff in the development of more specific GDC for the seismic qualification of Category I instrumentation and electric 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"

2. ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, "Requirements for Component Supports"

B. RDT Standards.

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

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

C. Guides.

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

3.10.4 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 3.10, "Seismic Qualification of Category I Instrumentation and Electrical Equipment," NUREG 75/087, September 1975
2. Branch Technical Position EICSB 10, "Electrical and Mechanical Equipment Seismic Qualification Program," ibid.

B. Pending Industry Standards and Codes.

1. ANSI N278.2, Draft, "Requirements for Functional Qualification of Power-Driven Valve Actuators for Safety-Related Applications in Nuclear Power Generating Stations"
2. ANSI N278.2.1, Draft, "Functional qualification of Valves for Safety-Related Applications in Nuclear Power Plants"
3. ANSI N278.2.2, Draft, "Functional qualifications of Power-Driven Valve Actuators for Safety-Related Applications in Nuclear Power Plants"
4. ANSI N278.2.3, Draft, "Special Qualifications for Safety-Related Pressure Relief and Self-Operated Check Valves for Nuclear Power Plants"
5. ANSI N278.2.4, Draft, "Functional Qualification Requirements for Power Operated Safety-Related Active Valve Assemblies for Nuclear Power Plants"
6. IEEE 535, Draft, "Qualification of Class IE Lead Storage Batteries for Nuclear Power Generating Stations"

3.10.5 Topics for Future Standards Development

None.

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

3.11.1 Environmental Design

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. For some equipment, there may be little or no change between normal and accident environmental conditions. Other equipment may undergo severe environmental transients during accidents. Herein, normal and accident environmental conditions are deemed to include all environmental conditions which may result from any normal or abnormal mode of plant operation, design basis events, post-design basis events, and containment tests.

Simply stated, the general requirements for environmental design and qualification are as follows:

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

Temperature, pressure, and humidity environments should be based on the most severe postulated accident affecting the particular component. The postulated environment should reflect the chemical composition of all fluids likely to affect a component (e.g., sodium, NaK, inert gas, smoke, caustic vapors, etc.).

The radiation environment for qualifying containment-related equipment should reflect source terms that are compatible with those postulated to arise from a DBA.

B. Equipment qualification should provide assurance that safety class systems will be capable of performing required safety functions under accident conditions for at least the required operating time. Subsequent equipment failure is allowable if it does not impact health and safety.

C. The equipment environmental capability should be demonstrated by appropriate testing, analysis, operating experience or a combination of these methods.

3.11.2 Applicable 10 CFR Criteria

The following General Design Criteria and other provisions of 10 CFR 50 were developed specifically for water-cooled reactors. The intent of these requirements should be considered by the reactor staff in the development of more specific criteria for the environmental design and qualification of safety-related mechanical and electrical equipment.

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

3.11.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes.

1. IEEE 317-1976 (ANSI N45.3), "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"
2. IEEE 323-1974 (ANSI N41.5), "Standards for Qualifying Class IE Equipment for Nuclear Power Generating Stations"
3. IEEE 334-1974 (ANSI N41.9), "Standard Type Test for Continuous Duty Class IE Motors for Nuclear Power Generating Stations"
4. IEEE 381-1977, "Criteria for Type Tests of Class IE Modules Used in Nuclear Power Generating Stations"
5. IEEE 382-1972 (ANSI N41.6), "Trial-Use Guide for Type Test of Class IE Electric Valve Operators for Nuclear Power Generating Stations"
6. IEEE 383-1974 (ANSI N41.10-1975), "Standard for Type Test of Class IE Electric Cables Field Splices, and Connections for Nuclear Power Generating Stations"
7. ANSI/ANS 58.3-1977 (N182), "Physical Protection for Systems and Components Important to Safety"

B. RDT Standards.

None.

C. Guides.

1. USNRC Regulatory Guide 1.40, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants"
2. USNRC Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants"

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

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

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

3.11.4 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 3.11, "Environmental Design of Mechanical and Electrical Equipment," NUREG 75/087, September 1975

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

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

B. Pending Industry Standards and Codes.

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

2. ANSI N687, (ANS 56.9), Proposed, "Environmental Envelopes to be Considered in Qualifying Safety-Related Equipment"

3. ANSI N688, Proposed, "Simultaneous Radiation Doses and Environmental Conditions to be Considered in Qualifying Safety-Related Equipment"

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

5. IEEE 535, Draft, "Qualification of Class IE Lead Storage Batteries for Nuclear Power Generating Stations"

3.11.5 Topics for Future Standards Development

None.

3.12 WELDING OF NUCLEAR COMPONENTS

3.12.1 Welding Guidelines

The documents listed in the following subsections contain guidance pertinent to the welding of mechanical components at reactor facilities. Guidance pertinent to the design of these components is included in other chapters of this Design Guide. Guidance pertinent to welding materials is found in Section 3.13.

3.12.2 Applicable 10 CFR Criteria

The following 10 CFR 50 Criteria were developed specifically for water-cooled reactors. The intent of these requirements should be considered by the reactor staff in the development of specific GDC for the welding of nuclear components.

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

3.12.3 Standards, Codes, and Guides

It is recommended that the following standards, codes, and guides be considered in the welding of nuclear components. Figure 3.12-1 is a guide to the application of these standards, codes, and guides.

A. Industry Standards and Codes.

- 1. ASME Boiler and Pressure Vessel Code, Section IX
- 2. AWS A3.0-76, "Welding Terms and Definitions"
- 3. ASW D1.1-75, "Structural Welding Code" (with Rev. 1-76)
- 4. AWS D10.0-69, "Standard for Qualification of Welding Procedures and Welders for Piping and Tubing"
- 5. AWS D12.1-75, "Reinforcing Steel Welding Code"
- 6. ANSI Z49.1-1973, "Safety in Welding and Cutting"

B. RDT Standards.

The following RDT standards modify or expand on requirements of the referenced ASME and AWS standards (listed in parentheses, below). The applicability of the additional requirements of the RDT standards should be considered on a case basis.

- 1. F6-2T, "Welding of Reactor Core Components and Test Assemblies"

Figure 3.12-1. Guide to Application of Welding Standards Codes, and Guides.

2. F6-5T, "Welding and Brazing Qualification," (Supplements ASME B&PV Code, Section IX)

75) 3. F6-6T, "Welding of Structural Components," (Supplements AWS D1.1-75)

C. Guides.

1. USNRC Regulatory Guide 1.19, "Nondestructive Examination of Primary Containment Liner Welds"

2. USNRC Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"

3. USNRC Regulatory Guide 1.34, "Control of Electroslag Weld Properties"

4. USNRC Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low Alloy Steel Components"

5. USNRC Regulatory Guide 1.50, "Control of Preheat Temperatures of Welding Low Alloy Steel"

6. USNRC Regulatory Guide 1.71, "Welding Qualification for Areas of Limited Accessibility"

3.12.4 Supplementary Information

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

A. Background Information.

1. Branch Technical Position MTEB 5-1, "Interim Position on Regulatory Guide 1.31, Control of Stainless Steel Welding," (Attached to Standard Review Plan 5.2.3, NUREG 75/087, September 1975)

B. Pending Industry Standards and Codes.

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

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. Guidance pertinent to the design of these components, systems, and structures is included in the following chapters.

3.13.2 Applicable 10 CFR Criteria

The following General Design Criterion and other provisions of 10 CFR 50 were developed specifically for water-cooled reactors. The intent of these requirements should be considered by the reactor staff in the development of more specific GDC for the selection and use of materials.

- A. GDC 1, "Quality Standards and Records"
- B. 10 CFR 50, Appendix G, "Fracture Toughness Requirements"
- C. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

3.13.3 Standards, Codes, and Guides

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

- A. Industry Standards and Codes.
 - 1. ASME Boiler and Pressure Vessel Code, Section II, "Material Specifications"
 - 2. ANSI N4.1-1973 (ASTM D 2953-72), "Classification System for Polymeric Materials for Services in Ionizing Radiation"
 - 3. ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities"
 - 4. ANSI N136-1973 (ASTM C 701-72), "Specification for Nuclear Grade Plutonium Metal"
 - 5. ANSI N137-1973 (ASTM C 710-72), "Specification for Plutonium Nitrate Solutions"
 - 6. ANSI N145-1973 (ASTM E 184-62), "Practice for Determining Effects of High Energy Radiation on the Mechanical Properties of Metallic Materials"
 - 7. ANSI N558-1974 (ASTM A 613-73), "Specification for Special Requirements for Steel Castings for Nuclear and Other Special Applications"

8. ANSI N560-1974 (ASTM A 652-73) "Specification for Special Requirements for Wrought Steel Welding Fittings for Nuclear and Other Special Applications"

9. ANSI N564-1974 (ASTM C 655-73), "Specification for Special Requirements for Pipe and Tubing for Nuclear and Other Special Applications"

10. ANSI N567-1974 (ASTM C 753-73), "Specification for Nuclear Grade, Sinterable Uranium Dioxide Powder"

11. ANSI N568-1975 (ASTM C 757-74a), "Specification for Nuclear Grade, Sinterable Plutonium Dioxide Powder"

12. ANSI N569 (ASTM C 750-74), "Specification for Nuclear Grade Boron Carbide Powder"

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

14. ANSI/ASTM A614-76, "Standard Specification for Bolting Material for Nuclear and Other Special Applications"

15. ANSI/ASTM A647-76, "Standard Specification for Special Requirements for Steel Plates for Nuclear and Other Special Applications"

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

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

18. ANSI/ASTM A654-76, "Standard Specification for Steel Forgings and Bars for Nuclear and Other Special Applications"

19. ASTM C859-78, "Definitions of Terms Relating to Nuclear Materials"

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

B. RDT Standards.

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

1. E-Series standards for liquid metal reactor core and reactivity control system material requirements.

2. M-Series standards for general material requirements.

C. Guides.

1. USNRC Regulatory Guide 1.36, "Non-Metallic Thermal Insulation for Austenitic Stainless Steel"
2. USNRC Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel"
3. USNRC Regulatory Guide 1.65, "Material and Inspections for Reactor Vessel Closure Studs"
4. USNRC Regulatory Guide 1.72, "Spray Pond Plastic Piping"
5. USNRC Regulatory Guide 1.85, "Code Case Acceptability," ASME, Section III, Materials
6. USNRC Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"
7. USNRC Regulatory Guide 1.136, "Material for Concrete Containments"

3.13.4 Supplementary Information

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

A. Background Information.

1. Branch Technical Position MTEB 5-1, "Interim Position on Regulatory Guide 1.31, Control of Stainless Steel Welding," (Attached to SRP 5.2.3), NUREG 75/087, September 1975
2. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," (Attached to SRP 5.3.2), *ibid.*
3. Branch Technical Position MTEB 5-5, "Fracture Toughness Requirements for ASME Code Class 2 and Class 3 Components," *ibid.*
4. "Building Materials Directory," Underwriters Laboratory, Inc., January 1977
5. MIL-F-51079B, "Filter Medium, Fire Resistant, High Efficiency," 1974

B. Pending Industry Standards and Codes.

1. ASTM P/N256, Draft, "Special Material Requirements for Liquid Metal Fast Breeder Reactors"

3.13.5 Topics for Future Standards Development

None.

3.14 DECOMMISSIONING OF REACTOR FACILITIES

3.14.1 Decommissioning Guidelines

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

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

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

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

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

3. Removal of Radioactive Components and Dismanteling - All radioactive materials having activities above unrestricted use activity levels (see USNRC Regulatory Guide 1.86) should 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 that will facilitate the eventual decommissioning of the facility and, in the process, will assist in maintaining occupational radiation exposure as low as reasonably achievable (see Chapter 12). 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 exposure 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 10 CFR Criteria

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

3.14.3 Standards, Codes, and Guides

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

A. Industry Standards and Codes.

None.

B. RDT Standards.

None.

C. Guides.

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

3.14.4 Supplementary Information

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

A. Background Information.

1. Sande, W.E., et al., "Decontamination and Decommissioning of Nuclear Facilities - A Literature Search," Battelle Pacific Northwest Laboratories, BNWL-1917, May 1975

2. AIF/NESP-009SR, "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives," Atomic Industrial Forum, November 1976

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

B. Pending Industry Standards and Codes.

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

3.14.5 Topics for Future Standards Development

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

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

CHAPTER 4

REACTOR

4.1 REACTOR CORE ASSEMBLY

4.1.1 Scope

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

4.1.2 Safety Function

The safety functions of the reactor core assembly and associated components are:

A. To 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. To form the primary barrier, by means of fuel element cladding, against the release of fission products from the fuel.

4.1.3 Definitions

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

- A. The fuel and fuel assemblies.
- B. Blanket and reflector assemblies.
- C. Reactivity control assemblies.
- D. Core support structures.
- E. Thermal shields.
- F. In-core experimental facilities (e.g., for test loops).

4.1.4 Design Guideline

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 likely to occur during its design lifetime. Fuel-cladding damage

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

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.

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, the core should maintain an in-place coolable geometry.

C. The core assembly should be designed to incorporate a suitable restraint system to ensure that no positive power coefficient can be introduced by core movement.

D. Reactor fuel subassembly pin spacing should ensure adequate coolant flow through the core considering the potential for reductions in coolant flow due to fuel swelling, fuel bowing, and fuel pin fouling.

E. Means should be provided to detect subassembly faults, to cope with these faults, and to protect against progressive subassembly fault propagation.

F. The reactor core should be designed to have a negative overall power coefficient in the power operating range, to provide a reliable feedback mechanism enhancing the stability of normal power operation and limiting reactivity excursions.

G. The potential effects of experimental facilities associated with a reactor core, including test loops and their contents, must 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.

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

I. 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.

J. The design of mechanical components of the core assembly should consider the guidance in Section 3.9. In particular, guidance in the ASME Code, Section III, Subsection NG, "Core Support Structures" should be considered.

K. The reactor core and the core support structure should be designed to Seismic Category I standards.

4.1.4.2 Maintenance, Testing, and Inspection

Inspection of reactor internals should consider the guidance of the ASME B&PV code, Section XI.

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. Primary Heat Transport 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, the intent of the following GDC that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of more specific GDC for 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 be considered in the design of the reactor core assembly:

- A. Industry Standards and Codes.
 - 1. ANSI N147-1973, "Practice for Examination of Fuel Element Cladding, Including the Determination of Mechanical Properties"
 - 2. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"
 - 3. ANSI N411-1975, "Nuclear Data Sets for Reactor Design Calculations"

4. ANSI N412-1975, (ANS 19.3), "The Determination of Neutron Reaction Rate Distributions and Reactivity of Nuclear Reactors"

5. ANSI/ANS 19.5-1978, "Requirements for Reference Reactor Physics Measurements"

6. ANSI/ANS 57.8-1978, "Fuel Assembly Identification"

B. RDT Standards.

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

1. E6-19T, "Radial Reflector Assembly for Liquid Metal Fast Breeder Reactors"

2. E13-5T, "FFTF Driver Fuel Pin"

3. E13-16T, "Driver Fuel assembly"

C. Guides.

None.

4.1.7 Supplementary Information

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

A. Background Information.

1. Olander, D.R., "Fundamental Aspects of the Performance of Fast Reactor Fuel," USDOE

2. "Advanced LMFBR Fuels," ANS Topical Meeting Proceedings, October 10-13, 1977, Tucson, Arizona

3. "Fast Reactor Fuel Element Technology," ANS Proceedings, April 13-15, 1971 Conference, New Orleans, Louisiana

4. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes.

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

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

3. ASME PTC 32.2, Draft, "Method for Measuring the Performance of Nuclear Reactor Fuel"

4.1.8 Topics for Future Standards Development

None.

4.2 REACTIVITY CONTROL SYSTEM

4.2.1 Scope

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

4.2.2 Safety Function

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

4.2.3 Definition

The reactivity control system is defined as the neutron absorbing or reflecting systems which must function in order to assure safe operation and shutdown of the reactor.

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 1 if they are responsible for accomplishing the reactor scram function to ensure that potential damage to the reactor core assembly and the reactor coolant system boundary do not exceed the guidelines in Section 4.1 and in Chapter 15 respectively.

2. Mechanical or fluid system portions of the reactivity control system that are part of the reactor coolant system boundary should be designed to the safety class guidelines applicable to the reactor coolant system (see Section 5.1).

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

3. Mechanical or fluid system portions of the reactivity control system that are not a part of the reactor coolant system pressure boundary and whose failure does not prevent safe shutdown of the reactor or cause damage to the reactor core assembly or reactor coolant system boundary in excess of the guidelines in Section 4.1 and in Chapter 15 respectively, may be designed to a suitable lower safety class considering the safety class guidelines in Table 3.2-1 and the consequence guidelines in Chapter 15.

B. Electrical portions of the reactivity control system required for safety should be designed to Class IE standards and should consider the guidelines for protection systems and Class IE electric systems discussed in Sections 7.1 and 8.1 respectively. Generally, these portions include only the parts of the reactivity control system required for responding to a scram signal from the protection system (see Section 7.1). Other electrical portions of the reactivity control system (e.g. the portions required only for normal reactivity control) may be Non-Class IE.

4.2.4.2 Guidelines for the Design of a Reactivity Control System

A. Two independent, diverse and functionally redundant reactor shutdown systems should be provided to satisfy the requirement that the reactor power level will be quickly and reliably reduced whenever plant conditions require such action.

B. Both reactivity control systems should be capable of shutting down an operating reactor and maintaining it subcritical in a hot condition. Only one of the reactivity control systems need be capable of holding the reactor subcritical in a cold condition with one reactivity control assembly in the position which results in the greatest insertion of positive reactivity.

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

D. The reactivity control system should be designed to "fail safe" on loss of power (electric, pneumatic, or hydraulic power, as appropriate). The normal reactivity control system does not require Class IE electric power. The protection system, which initiates a reactor scram, is supplied with Class IE electric power (see Section 7.1).

E. 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.

F. 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 for any plant operating condition.

G. Positive means should be provided, as applicable, to prevent rod motion (such as rod ejection) which could cause a reactivity change that results in a reactor power transient that is in excess of the capability to maintain adequate core cooling.

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

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

J. The design of 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 environmental conditions anticipated during normal, upset, emergency, and faulted events. 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 periodic functional tests of the entire system, including tests necessary to determine the actual reactivity worth of reactivity control assemblies, experimental fuel assemblies and experimental loops installed in the reactor core, as necessary, to ensure that the reactivity control system will be capable of completing its safety function, if required.

B. Testing provisions of Sections 7.1 and 8.1 apply to electrical portions of the reactivity control system required for safety.

C. Inspections of the reactivity control system should consider the guidance of the ASME B&PV code, Section XI.

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. Primary Heat Transport System.
- C. Protection System.
- D. Non-Safety I&C System.
- E. Class IE Electric Power System.

- F. Non-Class IE Electric Power System.
- G. Cooling Water System.
- H. Experimental Facilities.

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

4.2.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, the intent of the following GDC that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of more specific GDC for the design on the reactivity control system:

- A. 10 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

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

- A. Industry Standards and Codes.
 - 1. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"
- B. RDT Standards.

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

1. E6-5T, "Collapsible-Rotor, Roller Nut Control Rod Drive Mechanism for Sodium Service"

2. E6-25T, "Control Rod Absorber Pin for Liquid Metal Fast Reactors"

3. E6-33T, "Control Rod Assembly for Liquid Metal Fast Breeder Reactors"

C. Guides.

None.

4.2.7 Supplementary Information

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

A. Background Information.

1. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

4.2.8 Topics for Future Standards Development

None.

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CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED COOLANT SYSTEMS

5.1 PRIMARY HEAT TRANSPORT SYSTEM

5.1.1 Scope

This section provides guidelines for the design of a liquid metal primary heat transport system and for determining the minimum safety class of components of the system. Figure 5.1-1 is a general guide to the use of this section.

5.1.2 Safety Function

The safety functions of the primary heat transport system are:

A. To provide a barrier against the release of primary coolant and radioactivity to the environment.

B. To provide for adequate core cooling and thereby ensure that the fuel design guidelines of Section 4.1 are not exceeded during any facility operating condition.

5.1.3 Definition

The primary heat transport system is defined as those components which form the primary and intermediate liquid metal coolant system boundaries. The following components are generally included:

A. Reactor vessel.

B. Primary sodium system loop piping (loop type Category IV reactor facility only).

C. Primary sodium system pumps.

D. Intermediate sodium system loop piping.

E. Intermediate sodium system pumps.

F. Dump heat exchangers or steam generators (intermediate coolant side only), as appropriate.

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

5.1.4 Design Guidelines

5.1.4.1 Guidelines for Determining the Minimum Safety Class of the Primary Heat Transport System

A. Components which comprise part of the Primary Sodium Boundary (PSB) should be Safety Class 1. The PSB includes components such as the vessel, piping, pumps, tanks, and valves that are:

1. Part of the primary sodium system, or
2. Connected to the primary sodium system, up to and including any and all of the following:
 - a. The second of two valves normally closed or automatically isolable during normal reactor operation.
 - b. The passive barrier between the primary sodium system and the intermediate coolant loop.

B. Primary sodium system guard vessels should be Safety Class 2.

C. Those components which are extensions of the primary sodium boundary and are capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure may be Safety Class 3. If these lines penetrate the containment they should be designated to a safety class that is compatible with the containment (see Section 6.2).

D. Portions of the intermediate coolant loops inside and penetrating containment should be designed to Safety Class 2. Portions outside the containment that are provided with suitable containment isolation valves may be designed to Safety Class 3.

5.1.4.2 Guidelines for the Design of the Primary Heat Transport System (PHTS)

A. The PHTS (including the primary and intermediate coolant liquid metal loops) 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. Means should be provided to detect and control the effects of liquid metal leakage from the PHTS.

1. Guard vessels should be provided as necessary to maintain adequate coolant inventory in the PHTS and to ensure the availability of heat transfer paths to the residual heat removal system.

2. Equipment cells having PHTS components should have the capability of collecting potential liquid metal leakage and preventing fires or chemical reactions that may result from such leakage (see Section 9.4). Design basis

PHTS leaks should be postulated in order to determine the adequacy of the sodium fire protection subsystem.

C. Means should be provided to detect and control the effects of steam/water leakage from the secondary heat transport system into the intermediate liquid metal loops.

D. The intermediate liquid metal loops should be operated at a higher pressure than the primary sodium loop in order to minimize the likelihood of leakage of primary sodium into the intermediate loops.

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

F. The design of mechanical portions of the PHTS should consider the guidance in Section 3.9.

G. PHTS components should be environmentally qualified per the guidelines in Section 3.11.

H. The PHTS should be designed to facilitate the removal of residual heat from the core with a minimal dependence on active systems. When active systems are required for residual heat removal, redundant, and preferably diverse equipment should be used (see Section 5.3).

I. The minimum fracture toughness requirements for ferritic materials used in safety class pressure retaining components of PHTS should consider the guidance of 10 CFR 50, Appendix G. Requirements for other materials should be based on similarly conservative guidelines.

J. Instrumentation should be provided in the PHTS to assure that critical system parameters are maintained within safety limits.

K. Portions of the PHTS which have safety functions and which require electric power for operation should be capable of operation on both off-site and on-site power (assuming one is unavailable). Other portions need only be operable on off-site power.

L. The primary sodium coolant system should be designed to minimize the entrainment of gas in the coolant.

M. PHTS piping and components should be provided with heat tracing, as necessary, to prevent liquid metal coolant from freezing.

N. An inert cover gas (e.g., argon) should be provided for liquid metal systems.

O. Sodium and cover gas purity should be maintained by purification systems as described in Section 9.3.

P. In-vessel interim storage of spent fuel should be provided (see Section 9.2) to allow decay of the shortlived decay products and minimize the design requirements on the ex-vessel fuel storage facility.

5.1.4.3 Maintenance, Testing, and Inspection

A. The PHTS should be designed to facilitate in-service inspection during service life as required in the ASME B&PV Code, Section XI. Sufficient space should be available between the reactor vessel and the guard vessel to facilitate in-service inspection.

B. The PHTS should be designed to permit initial and periodic hydrostatic testing.

C. Reactor vessel material test specimens for a material surveillance program should be accommodated within the reactor coolant in the zone of maximum radiation exposure of the vessel. The irradiated test specimens shall be capable of removal and testing to establish the material properties of the irradiated reactor vessel. No material surveillance program is required for ferritic vessels which will be exposed to peak neutron fluences ($E > 1 \text{ MeV}$) $< 10^{17} \text{ n/cm}^2$ at the end of the design life. Requirements for material surveillance programs for nonferritic materials should be based on similarly conservative guidelines.

5.1.4.4 Interfaces With Other Systems

The PHTS may interface with other facility systems, including the following:

- A. Reactor Core Assembly.
- B. Reactivity Control System.
- C. Secondary Heat Transport System.
- D. Residual Heat Removal System.
- E. Reactor Building.
- F. Protection System.
- G. Non-Safety I&C Systems.
- H. Non-Class IE Electric Power System.
- I. Auxiliary Sodium and Purification Systems.
- J. Experimental Facilities.
- K. Radioactive Waste System.

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

5.1.5 Applicable 10 CFR Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, the intent of the following provisions of 10 CFR 50 that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of more specific GDC for the design of the Primary Heat Transport 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 31, "Fracture Prevention of Reactor Coolant Pressure Boundary"
- E. GDC 32, "Inspection of Reactor Coolant Pressure Boundary"
- F. GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment"
- G. 10 CFR 50.2(v), "Reactor Coolant Pressure Boundary"
- H. 10 CFR 50.55a, "Codes and Standards"
- I. 10 CFR 50, Appendix G, "Fracture Toughness Requirements"
- J. 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

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 the primary heat transport system:

- A. Industry Standards and Codes.
 - 1. ANSI N146-1973 (ASTM E185-73), "Surveillance Tests on Structural Materials in Nuclear Reactors"
 - 2. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"
- B. RDT Standards.

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

1. Al-5T, "Purity Requirements for Operating Sodium Reactor Systems"

C. Guides.

1. USNRC Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"

5.1.7 Supplementary Information

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

A. Background Information.

1. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna.

B. Pending Industry Standards.

1. ASTM P/N258, Draft, "Special Coolant Requirements for Liquid Metal Cooled Fast Breeder Reactors"
2. IEEE 622, Draft, "Recommended Practice for the Installation of Electric Pipe Heating Systems for Nuclear Power Generating Stations"

5.1.8 Topics for Future Standards Development

None.

5.2 SECONDARY HEAT TRANSPORT SYSTEM

5.2.1 Scope

This section provides guidelines for the design of secondary heat transport system components that are used to transfer heat from the primary heat transport system to the ultimate heat sink during normal operations. Not all Category IV reactors will have a secondary heat transport system. This section also provides guidelines for determining the minimum safety class of secondary coolant system components. Figure 5.2-1 is a guide to the use of this section.

5.2.2 Safety Function

The safety functions of the secondary heat transport system are those associated with any interfaces with the primary heat transport system (see Section 5.1) or a residual heat removal system (see Section 5.3). Other portions of the secondary heat transport system which do not interface with safety class systems generally do not have a safety function.

Figure 5.2-1. Guide to the Use of Section 5.2.

5.2.3 Definition

The secondary coolant system is defined as the water system which transfers heat from the primary heat transport system to an ultimate heat sink. This system may include the following:

- A. Steam generators (secondary side only).
- B. Main steam and feedwater systems.
- C. Turbine generator.
- D. Main condenser.
- E. Associated auxiliary systems (e.g., feed heaters, air ejectors, etc.).

5.2.4 Design Guidelines

5.2.4.1 Guidelines for Determining the Minimum Safety Class of the Secondary Heat Transport System

A. Portions of the steam generators interfacing with the intermediate liquid metal coolant loop (e.g., tubes and tube sheet) should be designed to a minimum of SC-3.

B. Portions of the secondary heat transport system required for residual heat removal should be designed to a minimum of SC-3.

C. Portions of the secondary heat transport system should be designed to a minimum of SC-3 if their failure may cause an off-site dose in excess of 0.5 rem to the whole body.

D. Other portions of the secondary heat transport system may be designed to non-nuclear standards.

5.2.4.2 Guidelines for the Design of a Secondary Heat Transport System

A. The secondary heat transport 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 Section 3.9).

B. Means should be provided to detect and limit the effects of water or steam leaks from the secondary heat transport system into the intermediate liquid metal loop. As a minimum, gross failure of a steam generator tube should be considered in the design of auxiliary equipment used to limit sodium/water reactions.

C. The materials of safety class pressure retaining components of the secondary heat transport system should have properties that afford protection

against brittle failure under all modes of plant operation, hydrostatic testing and accident conditions.

D. The design of mechanical components of the secondary heat transport system should consider the guidance in Section 3.9.

E. Safety class components of the secondary heat transport system should be environmentally qualified per the guidelines in Section 3.11.

F. Safety class portions of the secondary heat transport system should be Seismic Category I.

G. The secondary heat transport system should be provided with the capability of isolating leaking components.

H. Adequate overpressure protection for the secondary heat transport system should be provided.

I. Portions of the secondary heat transport system which have safety functions and which require electric power for operation should be capable of operating on both off-site and on-site power (assuming one is unavailable). Other portions need only be operable on off-site power.

5.2.4.3 Maintenance, Testing, and Inspection

A. Testing provisions should be incorporated to enable a periodic evaluation of the operability and functional performance of active components.

B. Portions of the secondary heat transport system interfacing with safety class systems should be designed to permit periodic testing of those safety class systems in order to verify their operational status.

C. Safety class portions of the secondary heat transport system should be designed to facilitate in-service inspection during service life.

5.2.4.4 Interfaces With Other Systems

The secondary heat transport system may interface with the following reactor facility systems:

A. Primary Heat Transport System.

B. Residual Heat Removal System.

C. Protection System.

D. Non-Safety I&C System.

E. Non-Class IE Electric Power System.

F. Ultimate Heat Sink System.

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

5.2.5 Applicable 10 CFR Criteria

None.

5.2.6 Standards, Codes, and Guides

None.

5.2.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 5.3.2.2, "Steam Generator Inservice Inspection," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 10.2, "Turbine Generator," *ibid.*
3. USNRC Standard Review Plan 10.2.3, "Turbine Disk Integrity," *ibid.*
4. USNRC Standard Review Plan 10.3, "Main Steam Supply System," *ibid.*
5. USNRC Standard Review Plan 10.4.1, "Main Condensers," *ibid.*
6. USNRC Standard Review Plan 10.4.2, "Main Condenser Evacuation System," *ibid.*
7. USNRC Standard Review Plan 10.4.3, "Turbine Gland Sealing System," *ibid.*
8. USNRC Standard Review Plan 10.4.4, "Turbine By-pass System," *ibid.*
9. USNRC Standard Review Plan 10.4.5, "Circulating Water Systems," *ibid.*
10. USNRC Standard Review Plan 10.4.6, "Condensate Cleanup System," *ibid.*
11. USNRC Standard Review Plan 10.4.7, "Condensate and Feedwater System," *ibid.*

12. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," (attached to SRP 3.6.1), *ibid.*

13. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," (attached to SRP 3.6.2), *ibid.*

14. Branch Technical Position MTEB 10-1, "Turbine Disk Integrity," *ibid.*

B. Pending Industry Standards and Codes.

None.

5.2.8 Topics for Future Standards and Development

None.

5.3 RESIDUAL HEAT REMOVAL SYSTEM

5.3.1 Scope

This section provides guidelines for determining the need for a residual heat removal (RHR) system that is provided to remove heat from the reactor coolant system following reactor shutdown. In addition, basic guidance relevant to the design of an RHR system is provided. Figure 5.3-1 is a guide to the use of this section.

5.3.2 Safety Function

The safety function of the 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 reactor fuel design guidelines and design guidelines for the primary heat transport system (PHTS) boundary specified in Sections 4.1 and 5.1 respectively are not exceeded during periods when long term shutdown cooling is required.

If these functions can be performed by entirely passive means (e.g., conductive heat transfer via reactor internal structures), no RHR is required.

5.3.3 Definition

The RHR system typically includes the following:

A. The heat exchangers used to remove residual heat from the PHTS.

B. The pumps, piping, and valves necessary to complete the fluid flow path for residual heat removal.

Figure 5.3-1. Guide to the Use of Section 5.3

5.3.4 Design Guidelines

5.3.4.1 Guidelines for Determining the Need for an RHR System

The residual heat removal system is required if heat must actively be removed from the primary heat transport system following a reactor shutdown in order to maintain the reactor core and the PHTS at acceptably low temperatures in order to prevent damage to the reactor fuel or the primary sodium boundary in excess of the guidelines in Sections 4.1 and 5.1 respectively. An RHR system may also be required in order to reduce reactor coolant temperature sufficiently to permit maintenance or refueling operations. Generally, Category IV reactors require an RHR system.

An RHR system should generally be designed to a minimum of Safety Class 3. Higher safety classes should be considered, as appropriate, to reflect safety class requirements of interfaces with the primary heat transport system (see Section 5.1) and the containment (see Section 6.2).

5.3.4.2 Guidelines for the Design of an RHR System

A. At least two independent, diverse and functionally redundant decay heat removal systems should be provided.

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

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

D. The RHR may be manually actuated if sufficient time is available for operator response (see Chapter 15). Manual actuation circuitry should consider the guidelines for protection systems in Section 7.1. If automatic RHR system actuation is provided, it too should consider the guidelines in Section 7.1 and should be provided with a manual actuation backup.

E. Portions of the RHR system instrumentation and controls associated with the protection system should meet the guidelines specified in Section 7.1.

F. The RHR system should be capable of operating on both off-site and on-site power (assuming that one is unavailable). RHR system pumps and motor operated valves should be provided with Class IE power. RHR system liquid metal lines that are not part of a normal heat transport path should be provided with heat tracing that is supplied with Class IE electric power to ensure the availability of the RHR system when required.

G. The RHR should be designed as a Seismic Category I system.

H. The design of mechanical portions of the RHR system should consider the guidance in Section 3.9.

I. The RHR system should be protected from the effects of missiles and the dynamic effects of pipe rupture as described in Sections 3.5 and 3.6 respectively.

J. Safety class portions of the RHR system should be environmentally qualified to withstand the post-accident environment within the reactor building (see Section 3.11).

5.3.4.3 Maintenance, Testing, and Inspection

A. The RHR system should be designed to permit operational testing of the entire system during periods when the reactor is shut down. This testing should verify the performance capability of the RHR system.

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

C. The RHR system should be designed to permit periodic inservice inspection of important components such as heat exchangers, pumps, valves, and piping as required by the ASME Code, Section XI.

5.3.4.4 Interfaces With Other Systems

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

- A. Primary Heat Transport System.
- B. The Reactor Building.
- C. Protection System.
- D. Class IE Electric Power System.
- E. Cooling Water System.
- F. Ultimate Heat Sink.

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

5.3.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, the intent of the following criterion that was developed specifically for water-cooled reactors should be considered by the reactor staff in the development of more specific criteria for the design of a residual heat removal system:

A. GDC 34, "Residual Heat Removal"

5.3.6 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 residual heat removal system:

A. Industry Standards and Codes.

1. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"

2. IEEE 622, Draft, "Recommended Practice for the Installation of Electric Pipe Heating Systems for Nuclear Power Generating Stations"

B. RDT Standards.

None.

C. Guides.

None.

5.3.7 Supplementary Information

None.

5.3.8 Topics for Future Standards Development

None.

CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 EMERGENCY CORE COOLING SYSTEM

A Category IV reactor does not generally require a safety class liquid metal makeup system in order to maintain adequate coolant inventory in the primary heat transport system following a postulated break in the system boundary. Adequate coolant inventory is maintained by safety class guard vessels described in Section 5.1. Adequate post-accident heat removal is provided by the residual heat removal system described in Section 5.3

6.2 REACTOR BUILDING

6.2.1 Scope

This section provides guidelines for the design of a containment for a Category IV reactor facility. Figure 6.2-1 is a guide to the use of this section.

Confinement-type reactor buildings are less effective than containments in controlling the release of radioactivity to the environment following postulated accidents. Safety-related and non-safety-related reactor buildings do not control the release of radioactivity. Reactor buildings other than containments are generally not suitable for use at a Category IV reactor facility. If it should be determined that a particular Category IV reactor facility can remain within the consequence guidelines in Chapter 15 by using a reactor building other than a containment, that facility should consult the Design Guide for Category II Reactors, Light and Heavy Water-Cooled Reactors, for appropriate design guidelines.

6.2.2 Safety Function

A. A containment should 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. A containment should provide protection against the effects of severe natural phenomena, missiles and pipe whip to enclosed safety class components and systems.

6.2.3 Definition

A containment is defined as those structures and components (including the pressure retaining containment shell, isolation valves, heat removal systems, and cleanup systems) which enclose the reactor and associated systems and must

Figure 6.2-1. Guide to the Use of Section 6.2.

function in order to provide a low leakage, pressure retaining barrier against the uncontrolled release of radioactivity to the environment and which additionally protects enclosed systems against the effects of severe natural phenomena, missiles, and pipe whip.

6.2.4 Design Guidelines

6.2.4.1 Guidelines for Determining the Minimum Safety Class of a Containment

A Category IV reactor facility containment should generally be designed to Safety Class 2 standards.

6.2.4.2 Guidelines for the Design of a Containment and Associated Systems

A. The Containment Structure.

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

- a. The spectrum of postulated accidents (see Chapter 15).
- b. Reactor core decay heat.
- c. Residual heat sources (e.g., primary or intermediate sodium heat transport system components).
- d. The operation of the containment cooling system (if provided).
- e. The operation of the core residual heat removal system.
- f. Other potential energy sources (e.g., sodium fires and combustible gas reactions).
- g. Uncertainties that may necessitate additional design margin.

2. The containment integrity should be maintained for normal, upset, emergency, and faulted operating conditions (defined in Section 3.9).

3. The containment should be protected against potential effects of postulated Design Basis Accidents (see Chapter 15), as necessary, to ensure that the accident dose guidelines in Chapter 15 are not exceeded.

4. The containment should be protected from the effects of sodium releases to equipment cells, particularly those cells containing the primary heat transport system equipment. Protection for these cells may be provided by the following measures:

- a. Increased cell structural capability (e.g., higher design pressure) to withstand pressure transients that may result from sodium being dispersed in the cell.

- b. Cell venting.
- c. Other suitable means.

5. Containment design features should minimize the possibility and extent of hydrogen producing reactions (e.g., sodium-concrete reactions) to an acceptable level. These design features may include the following:

- a. Steel lined concrete cells.
- b. Cell configuration to minimize the spread of released sodium.

6. A hydrogen recombiner system or containment atmosphere dilution system should be provided, if necessary, to prevent post-accident hydrogen concentrations from reaching explosive limits thus threatening containment integrity.

7. The containment and associated systems required for maintaining the containment safety function should be designed to Seismic Category I standards (see Section 3.8).

8. The design of mechanical components of the containment and associated systems should consider the design guidelines in Section 3.9.

9. The potential releases of radioactivity to the containment, arising from a spectrum of postulated accidents, should be considered in establishing the maximum containment leak rate. (See Chapter 15 for accidents to be considered.)

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

11. The containment structure should provide protection to enclosed structures from the effects of severe natural phenomena, missiles, and pipe whip (see Chapter 3).

12. The safety class of the systems associated with the containment (containment isolation system, containment heat removal system, containment cleanup system, and hydrogen recombiner system) should be compatible with the safety class of the containment.

13. Provisions for containment purging or ventilation should be included, if necessary, for improving working conditions, reducing occupational exposure to airborne radioactivity, and reducing hydrogen concentration within the containment. On-site and off-site dose and effluent concentration guidelines to be considered in the design and operation of containment venting and purging equipment are discussed in Chapter 12.

B. The Containment Isolation Systems.

1. Means should be provided to isolate the containment following postulated accidents which may release radioactivity to the containment atmosphere. Isolation of the containment will prevent or minimize the release of radioactivity to the environment through fluid system lines (including vent or purge lines) penetrating containment.

2. When active components are provided to effect containment isolation, a single active failure in the isolation system (e.g., failure of one valve to close) should not result in a failure to completely isolate the containment, unless the radiological consequences of such a failure are within the guidelines for the appropriate facility operating condition (see Chapter 15) (e.g., as may be the case with small instrument lines). Application of the single-failure criterion to system design is discussed in Chapter 15.

3. Adequate instrumentation should be provided in order to actuate isolation valves when necessary, or to notify the operator of conditions requiring manual isolation of the containment.

4. Instrumentation required to manually or automatically initiate containment isolation should meet the guidelines for protection systems listed in Section 7.1.

5. The potential releases of radioactivity to the containment should be considered in determining the timing requirements for containment isolation valves in order to assure radiological consequences of postulated accidents are within the guidelines of the appropriate facility operating condition (see Chapter 15).

6. The containment isolation system components should be capable of performing their safety function under the environmental conditions specified in Section 3.11.

C. The Containment Heat Removal System.

1. A containment heat removal system should be designed to have a sufficient heat removal capacity to prevent exceeding reactor containment design pressure and temperature limits for all postulated accidents inside containment.

2. This system is not necessary if post-accident heat can adequately be dissipated to the containment and other contained passive heat sinks.

3. When required for safety, the containment heat removal system should incorporate sufficient redundancy to complete its safety function with the addition of a single failure in the containment heat removal system. Application of the single-failure criterion to system design is discussed in Chapter 15.

4. Adequate instrumentation should be provided in order to actuate a containment heat removal system, when necessary, or to notify the operator of conditions requiring manual initiation of containment heat removal.

5. Instrumentation required for automatic or manual actuation of the containment heat removal system should meet the guidelines for protection systems in Section 7.1.

D. The Containment Cleanup System.

1. The containment cleanup system (a recirculating system within the containment) should be designed to facilitate the removal of airborne radioactive material available for leakage to the environment following its release to the containment during a postulated accident (see Chapter 15).

2. This system is not required if the conservatively calculated maximum postulated off-site and on-site doses can be maintained within the guidelines for the applicable facility operating condition (see Chapter 15) without containment cleanup.

3. When required for safety, the containment cleanup system should incorporate sufficient redundancy to complete its safety function with the addition of a single failure in the system. Application of the single-failure criterion to system design is discussed in Chapter 15.

4. The design of an atmosphere cleanup system should consider the guidelines of 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 (activated charcoal, metal zeolite or equivalent adsorbent). HEPA filters should be provided after the adsorber bed, if necessary, to remove additional particulates (e.g., charcoal or zeolite fines) from the exhaust stream.

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

A. A containment should be designed to permit periodic integrated leak rate testing.

B. The performance of periodic functional tests of the complete containment and associated systems should be possible during periods when the reactor is shut down.

C. 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.

6.2.4.4 Interfaces With Other Systems

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

- A. Primary Heat Transport System.
- B. Residual Heat Removal System.
- C. ESF Ventilation System.
- D. Protection System.
- E. Non-Safety I&C System.
- F. Class IE Electric Power System.
- G. Non-Class IE Electric Power System.
- H. Cooling Water System.
- I. Fuel-Handling and Storage System.
- J. Fire Protection System.
- K. Experimental Facilities.
- L. Radioactive Waste System.
- M. 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 10 CFR Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, the intent of the following provisions of 10 CFR 50 that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for the design of containments:

- A. GDC 16, "Containment Design"
- B. GDC 38, "Containment Heat Removal"
- C. GDC 39, "Inspection of Containment Heat Removal System"
- D. GDC 40, "Testing of Containment Heat Removal System"
- E. GDC 41, "Containment Atmosphere Cleanup"

- F. GDC 42, "Inspection of Containment Atmosphere Cleanup System"
- G. GDC 43, "Testing of Containment Atmosphere Cleanup System"
- H. GDC 50, "Containment Design Basis"
- I. GDC 51, "Fracture Prevention of Containment Pressure Boundary"
- J. GDC 52, "Capability for Containment Leak Rate Testing"
- K. GDC 53, "Provisions for Containment Testing and Inspection"
- L. GDC 54, "Piping Systems Penetrating Containment"
- M. GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment"
- N. GDC 56, "Primary Containment Isolation"
- O. GDC 57, "Closed System Isolation Valves"
- P. GDC 60, "Control of Releases of Radioactive Materials to the Environment"
- Q. GDC 64, "Monitoring Radioactivity Releases"
- R. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"

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 containments:

- A. Industry Standards and Codes.
 - 1. ANSI N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors"
 - 2. ANSI N101.2-1972, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities"
 - 3. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"
 - 4. ANSI N271-1976 (ANS 56.2), "Containment Isolation Provisions for Fluid Systems"
 - 5. ANSI/ASME N509-1976, "Nuclear Power Plant Air Cleaning Units and Components"

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

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

8. IEEE 317-1976, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"

B. RDT Standards.

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

1. E9-1T, "HEPA Filters"

2. E10-8T, "Steel Containment Vessel"

C. Guides.

1. USNRC Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment"

2. 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"

3. USNRC Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants"

4. 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"

5. USNRC Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems"

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

6.2.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 6.2.1, "Containment Functional Design," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 6.2.2, "Containment Heat Removal Systems," *ibid.*
3. USNRC Standard Review Plan 6.2.3, "Secondary Containment Functional Design," *ibid.*
4. USNRC Standard Review Plan 6.2.4, "Containment Isolation System," *ibid.*
5. USNRC Standard Review Plan 6.2.5, "Combustible Gas Control in Containment," *ibid.*
6. USNRC Standard Review Plan 6.2.6, "Containment Leakage Testing," *ibid.*
7. USNRC Standard Review Plan 6.5.1, "ESF Filter Systems," *ibid.*
8. USNRC Standard Review Plan 6.5.3, "Fission Product Control Systems," *ibid.*
9. Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants" (attached to SRP 6.2.3), *ibid.*
10. Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations," (attached to SRP 6.2.4), *ibid.*
11. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.
12. ORNL-NSIC-26, "Testing of Containment Systems Used with Light Water-Cooled Power Reactors," Oak Ridge National Laboratory, August 1968
13. Burchsted, C.A., et al., "Nuclear Air Cleaning Handbook," ERDA 76-21, Oak Ridge National Laboratory, March 31, 1976.
14. MIL-F-51068D, "Filter Particulate High Efficiency Fire Resistant"
15. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes.

1. ANSI N274 (ANS 56.8), Draft, "Containment System Leakage Testing Requirements"
2. ANS 54.3, Proposed, "Principal Design Criteria for LMFBR Containments"
3. AACC CS-8T, Draft, "Tentative Standard for High Efficiency Gas Phase Adsorber Cells," American Association of Contamination Control"

6.2.8 Topics for Future Standards Development

None.

6.3 ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

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

The safety functions of the ESF ventilation system are to:

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

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

6.3.3 Definition

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

6.3.4 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.

CONTINUED ON NEXT PAGE

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

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

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

For operating and supervisory personnel responsible for reactor safety, the calculated maximum doses following postulated accidents may be based on evacuation commencing after the reactor has been placed in a safe shutdown condition. The calculated maximum doses to other on-site personnel may be based on immediate evacuation.

2. Toxic Gas.

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

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

C. If required for safety, 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 ESF ventilation system should be Seismic Category I.

B. The design of mechanical components of the ESF ventilation system should consider the guidance in Section 3.9.

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

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

E. The selection of the type of 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 and 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.

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

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

I. The design of an ESF ventilation system should consider the guidelines in USNRC Regulatory Guide 1.52. Demisters preceeding 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 (activated charcoal, metal zeolite or equivalent adsorber). HEPA filters, after the iodine adsorber bed, should be provided, if necessary, to remove additional particulates (e.g., charcoal or zeolite fines) from the exhaust stream.

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

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 the 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.

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. Cooling Water System.
- E. Fuel-Handling and Storage System.
- F. Fire Protection System.
- G. Radioactive Waste System.
- H. 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 feature 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 N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"
 - 3. ANSI/ASME N509-1976, "Nuclear Power Plant Air Cleaning Units and Components"
 - 4. ANSI N510-1975, "Testing of Nuclear Air Cleaning Systems"
 - 5. ANSI/UL 586-1977, "Safety Standard for Test Performance of High-Efficiency Particulate Air Filter Units"

6. ANSI B193.1-76, (ASHRAE 55-74), "Thermal Environmental Conditions for Human Occupancy"

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

A. Background Information.

1. USNRC Standard Review Plan 6.4, "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. Griwatz, G.H., Friel, J.V., and Bicehouse, J.L., "Entrained Moisture Separators for Fine (1-10) Water-Air-Steam Service: Their Performance, Development, and Status," MSAR 71-45, Mine Safety Appliances Research Corporation

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

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

B. Pending Industry Standards and Codes.

1. ANSI N189 (ANS 59.2), Proposed, "HVAC Systems Important to Safety Located Outside Primary Containment"

6.3.8 Topics for Future Standards Development

None

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 is to 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 safety-related instrumentation is to 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.

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

7.1.4 Design Guidelines

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

A protection system should be provided if automatically or manually initiated protective actions must be taken in order to assure safe reactor shut-down and cooling or to mitigate the consequences of postulated accidents. Guidelines for determining consequence limits for postulated accidents are found in the following sections:

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

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, Chapters 12 and 15).

E. Off-site radiological consequences guidelines (Chapters 12 and 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. Fluid system and mechanical components in the protection system that are part of the primary heat transport system should be designed to the same minimum safety class as the sodium loop components (see Section 5.1). Other fluid system and mechanical components associated with engineered safety feature systems should be designed to the same minimum safety class as the ESF system.

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 shut down 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, functional diversity, and equipment diversity to assure that the protection system safety functions can be completed, when required, during a postulated accident with the addition of a single failure in the protection system. 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, light emitting and photodiodes, or other suitable qualified isolation devices) from non-safety instrumentation and control systems which may receive protection system output signals so that the operation or 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 during normal, accident, and post-accident conditions.
3. The need for safety-related operator actions during accident and post-accident conditions.

J. Protection system instrumentation and controls sufficient for placing the 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. As a minimum, this should include the ability to manually 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. The protection system 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 the protection system can operate satisfactorily on Non-Class IE electric power and 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 would generally not apply to Category IV reactor facilities.

O. The protective system should be qualified for all environmental conditions anticipated during normal operation and postulated accidents (see Section 3.11) in which the system is required to perform its safety function.

P. The protection system should be protected from the effects of severe natural phenomena, missiles, and pipe whip 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 and calibration of protection system instrument channels. On-line tests should not prevent the protection system from performing its safety functions.

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

D. Test intervals and allowable repair times for protection systems should be determined such that system malfunctions are detected and repaired in a timely manner which assures that the system meets desired availability goals.

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. Primary Heat Transport System.

- D. Secondary Heat Transport System.
- E. Residual Heat Removal System.
- F. Reactor Building.
- G. ESF Ventilation System.
- H. Non-Safety I&C System.
- I. Class IE Electric Power System.
- J. Fuel-Handling and Storage 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, the intent of the following provisions of 10 CFR 50 that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for the design of a protection system:

- A. GDC 10, "Reactor Design"
- B. GDC 12, "Suppression of Reactor Power Oscillations"
- C. GDC 13, "Instrumentation and Controls"
- D. GDC 15, "Reactor Coolant System Design"
- E. GDC 19, "Control Room"
- F. GDC 20, "Protection System Functions"
- G. GDC 21, "Protection System Reliability and Testability"
- H. GDC 22, "Protection System Independence"
- I. GDC 23, "Protection System Failure Modes"
- J. GDC 24, "Separation of Protection and Control Systems"
- K. GDC 25, "Protection System Requirements for Reactivity Control Malfunctions"

- L. GDC 29, "Protection Against Anticipated Operational Occurrence"
- M. GDC 63, "Monitoring Fuel and Waste Storage"
- N. 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 317-1976 (ANSI N45.3), "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"
3. IEEE 336-1977 (ANSI N45.2), "Installation Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
4. IEEE 338-1977, "Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems"
5. IEEE 352-1975 (ANSI N41.4-1976), "General Principles for Reliability Analysis of Nuclear Power Generating Stations"
6. IEEE 380-1975, "Definition of Terms Used in IEEE Nuclear Power Generating Station Standards"
7. IEEE 384-1977 (ANSI N41.14), "Criteria for Independence of Class IE Equipment and Circuits"
8. IEEE 420-1973 (ANSI N41.17), "Class IE Control Switchboards for Nuclear Power Generating Stations"
9. IEEE 494-1974, "Method for Identification of Documents Related to Class IE Equipment and Subsystems for Nuclear Power Generating Stations"
10. IEEE 497-1977, "Trial-Use Criteria for Post-Accident Monitoring Instrumentation for Nuclear Power Generating Stations"
11. IEEE 500-1977, "Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear Power Generating Stations"
12. IEEE 566-1977, "Recommended Practice for Design of Display and Control Facilities for Control Rooms of Nuclear Power Generating Stations".

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

14. IEEE 603-1977, "Trial Use Standard Criteria for Safety Systems for Nuclear Power Generating Stations"

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

16. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"

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

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

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

4. C16-3T, "PPS Buffers"

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

6. Other RDT C-Series standards for protection system sensors for liquid metal service.

C. Guides.

1. USNRC Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment"

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

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

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

5. USNRC Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants"

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

7. USNRC Regulatory Guide 1.97, "Instrumentation for Light Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident"

8. USNRC Regulatory Guide 1.105, "Instrument Spands and Setpoints"

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

7.1.7 Supplementary Information

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

A. Background Information.

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

2. USNRC Standard Review Plan 7.3, "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 EISCB 24, "Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times," *ibid.*

9. Branch Technical Position EISCB 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 1978

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. Hirsch, H.M., "Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems," NEDO-10739, General Electric, January 1973

14. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice, International Atomic Energy Agency, Vienna

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

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

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

B. Pending Industry Standards and Codes.

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

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

3. 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.

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

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 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 interconnection with a protection system should be designed to provide adequate electrical separation and independence between the two systems (e.g., via protection system buffer amplifiers or other suitable isolation devices that are part of the protection system) so that the operation or failure of the non-safety Class I&C system will not degrade the ability of the protection system to perform its safety function when required.

D. Physical separation or protection with barriers should be utilized as a means of assuring that failures in a non-safety I&C system will not prevent a protection system from completing its safety function when required.

E. Non-safety I&C systems need not be qualified for the environmental conditions resulting from severe natural phenomena or postulated accidents.

7.2.4.2 Maintenance, Testing, and Inspection

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

7.2.4.3 Interfaces With Other Systems

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

- A. Reactivity Control System.
- B. Primary Heat Transport System.
- C. Secondary Heat Transport System.

- D. Reactor Building.
- E. Protection System.
- F. Non-Class IE Electric Power System.
- G. Fuel-Handling and Storage System.
- H. Auxiliary Sodium and Purification Systems.
- I. Fire Protection System.
- J. Experimental Facilities.
- K. 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, the intent of the following GDC that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for the design of non-safety instrumentation and control systems:

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

7.2.6 Standards, Codes, and Guides

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

- A. Industrial Standards and Codes.
 - 1. ANSI C1-1975 (NFPA 70-1975), "National Electrical Code"
 - 2. ANSI C2-1977, "National Electrical Safety Code"
 - 3. ANSI C19 (Series), "Industrial Control Apparatus"
 - 4. ANSI N413-1974, "Guidelines for the Documentation of Digital Computer Programs"
 - 5. IEEE 317-1976 (ANSI N45.3), "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"

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

7. "Standards and Practices for Instrumentation" 5th Edition, Instrument Society of America, 1977

8. NEMA IS 1.1, "Enclosures for Industrial Controls"

9. NEMA ICS-1970, "NEMA Standards Publication, Industrial Controls and Systems"

10. 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"

4. Other RDT C-series standards for non-safety I&C system sensors.

C. Guides.

1. USNRC Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants"

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

7.2.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 7.7, "Control Systems Not Required for Safety," NUREG 75/087, September 1975

2. Harrer, J.M., and Beckerly, J.G., "Nuclear Power Reactor Instrumentation Systems 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 the design of a Class IE electric power 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 system safety functions when required. This system may utilize an on-site electric generation capability to provide power to safety class systems when off-site power is not available.

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. Standby ac and dc 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 nonessential 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 supplied with IE electric power generally include those that are required for safe reactor shutdown and cooling and for preventing an uncontrolled release of radioactivity to the environment with dose consequences in excess of the guidelines in Chapter 15.

A Class IE electric power system may not be required if loss of all electric power to the protection system and to safety class fluid and mechanical systems places the reactor in a safe shutdown condition. Additionally, no active electric-powered systems should be 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. It should additionally be

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

demonstrated that the protection system can operate satisfactorily on Non-Class IE quality electric power.

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 capacity and capability is available during a postulated accident with the addition of a single failure 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 assure their disconnection during emergencies.

F. The Class IE electric power system should be protected from the effects of severe natural phenomena, missiles, and pipe whip 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. Reactivity Control System.
- B. Residual Heat Removal System.
- C. Reactor Building.
- D. ESF Ventilation System.

- E. Protection System.
- F. Cooling Water System.
- G. Radiation 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, the intent of the following GDC and other provisions of 10 CFR 50 that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for 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 308-1074 (ANSI N41.12-1975), "Class IE Power Systems for Nuclear Power Generating Stations"

3. IEEE 317-1976 (ANSI N45.3), "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"
4. IEEE 334-1974 (ANSI N41.9), "Type Test of Continuous Duty Class IE Motors for Nuclear Power Generating Stations"
5. 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"
6. IEEE 338-1975, "Criteria for Periodic Testing of Nuclear Power Generating Station Protection Systems"
7. IEEE 380-1975, "Definition of Terms Used in IEEE Nuclear Power Generating Station Standards"
8. IEEE 382-1972 (ANSI N41.6), "Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations"
9. IEEE 383-1974 (ANSI N41.10-1975), "Type Test of Class IE Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations"
10. IEEE 384-1977 (ANSI N41.14), "Criteria for Independence of Class IE Equipment and Circuits"
11. IEEE 387-1977 (ANSI N41.13), "Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations"
12. IEEE 420-1973 (ANSI N41.17), "Class IE Control Switchboards for Nuclear Power Generating Stations"
13. IEEE 422-1977, "Guide for the Design and Installation of Cable Systems in Power Generating Stations"
14. IEEE 446-1974, "Recommended Practice for Emergency and Standby Power Systems"
15. IEEE 450-1975 (ANSI N41.15-1976), "Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations"
16. IEEE 484-1975 (ANSI N41.24-1976), "Recommended Practices for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations"
17. IEEE 494-1974, "Method for Identification of Documents Related to Class IE Equipment and Systems for Nuclear Power Generating Stations"
18. IEEE 500-1977, "Guide to the Collection and Presentation of Electrical, Electronic and Sensing Component Reliability Data for Nuclear Power Generating Stations"

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

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

21. IEEE 603-1977, "Trial Use Standard Criteria for Safety Systems for Nuclear Power Generating Stations"

22. ANSI N195-1976, (ANS 59.51), "Fuel Oil Systems for Standby Diesel Generators"

23. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"

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 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies"

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

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

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

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

7. USNRC Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants"

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

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

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

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

12. USNRC Regulatory Guide 1.108, "Periodic Testing of Diesel Generators Used as On-Site Electric Power Systems at Nuclear Power Plants"

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

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

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

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

17. USNRC Regulatory Guide 1.137, "Fuel Oil Systems for Standby Diesel Generators"

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 that are distributed on only 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 (On-Site)," *ibid.*

3. USNRC Standard Review Plan 8.3.2, "DC Power Systems (On-Site)," *ibid.*

4. Branch Technical Position EICSB 2, "Diesel-Generator Reliability Qualification Testing," *ibid.*

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

6. Branch Technical Position EICSB 8, "Use of Diesel-Generator Sets for Peaking," *ibid.*

7. Branch Technical Position EICSB 11, "Stability of Off-Site Power System," *ibid.*

8. Branch Technical Position EICSB 17, "Diesel Generator Protective Trip Circuit Bypasses," *ibid*.

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

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

11. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

12. IAEA Safety Guide SG-D7, Draft, "Safety-Related Electric Power Systems"

13. IAEA Safety Guide SG-D7A, Draft, "Emergency Electric Power Systems"

B. Pending Industry Standards and Codes.

1. ANSI N709 (ANS 59.52), Proposed, "Lubricating Oil Systems for Standby Diesel Generators"

2. ANSI N710 (ANS 59.53), Proposed, "Starting Air Systems for Standby Diesel Generators"

3. ANSI N711 (ANS 59.54), Proposed, "Combustion Air Systems for Standby Diesel Generators"

4. ANSI N712 (ANS 59.55), Proposed, "Coolant System for Standby Diesel Generators"

8.1.8 Topics for Future Standards Development

None.

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.

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

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 will not prevent a Class IE system from completing its safety function when required.

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

8.2.4.2 Maintenance, Testing, and Inspection

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

8.2.4.3 Interfaces With Other Systems

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

- A. Reactivity Control System.
- B. Primary Heat Transport System.
- C. Secondary Heat Transport System.
- D. Non-Safety Instrumentation and Control Systems.
- E. Cooling Water Systems.
- F. Fuel-Handling and Storage System.
- G. Auxiliary Sodium and Purification Systems.
- H. Fire Protection System.

I. Experimental Facilities.

J. Radioactive Waste System.

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

8.2.5 Applicable General Design Criteria

In addition to the 10 CFR 50, Appendix A, General Design Criteria listed in Chapter 3, the intent of the following GDC that were developed specifically for water-cooled reactors should be considered by the reactor staff in developing GDC for 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 67-1972 (ANSI C50.30-1972), "Guide to the Operation and Maintenance of Turbine Generators"

5. IEEE 141-1976, "Electric Power Distribution for Industrial Plants"

6. IEEE 142-1972 (ANSI C114.1-1973), "Recommended Practice for Grounding Industrial and Commercial Power Systems"

7. IEEE 317-1976 (ANSI N45.3), "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"

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

B. RDT Standards.

Guide.

C. Guides.

1. USNRC Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants"

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

8.2.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 8.2, "Off-Site Power System," NUREG 75/087, September 1975

2. Branch Technical Position EICSB 11, "Stability of Off-Site Power Systems," (in SRP Appendix 7A), *ibid*.

B. Pending Industry Standards and Codes.

None.

8.2.8 Topics for Future Standards Development

None.

CHAPTER 9

AUXILIARY SYSTEMS

9.1 COOLING WATER SYSTEMS AND THE ULTIMATE HEAT SINK

9.1.1 Scope

This section provides guidelines for the design of cooling water systems responsible for transferring heat from structures, systems, and components 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

The safety function of a cooling water system and ultimate heat sink is to transfer the heat load necessary to support the operation of safety class components, systems, and structures during normal operation and accident conditions. A cooling water system and ultimate heat sink have no safety function if they are not required to support the operation of safety class components, systems, or structures.

9.1.3 Definition

The cooling water systems include all components that form an open or closed cooling loop(s) that transfers heat from heat exchangers and components in the facility to 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. Cooling water systems or portions of cooling water systems are generally Safety Class 3 if they are required for cooling safety class systems or structures or if their failure could result in off-site doses in excess of 0.5 rem.

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

C. Other portions of the cooling water system may be non-nuclear if their failure will not result in off-site doses in excess of 0.5 rem.

9.1.4.2 Guidelines for the Design of a Cooling Water System

A. Safety class cooling water systems should be designed to incorporate sufficient redundancy and independence to ensure that safety class systems are

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

adequately cooled during normal operations and accident conditions with the addition of a single failure of a component in the cooling water system. Application of the single-failure criterion to system design is discussed in Chapter 15.

B. 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 accident condition.

C. Portions of a cooling water system that serve liquid-metal-containing systems or components should be provided with means for detecting and limiting the effects of leakage between a liquid metal system and the cooling water system. An alternative approach for certain applications would be to provide a separate liquid metal component cooling system which uses a coolant other than water to minimize potential reactions with liquid metal. The functional requirements for this alternative system are the same as those discussed in this section for the cooling water system.

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

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

F. A safety class cooling water system should be operable on off-site and on-site power (assuming one is unavailable). Non-safety portions of cooling water systems need only be operable on off-site power.

G. A safety class cooling water system should be Seismic Category I. Non-safety portions of cooling water systems are generally Non-Seismic Category I.

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

I. The safety class cooling water systems and the ultimate heat sink should be protected against the effects of severe natural phenomena and missiles (see Chapter 3).

J. A safety class cooling water system should be environmentally qualified as per the guidelines in Section 3.11.

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.

C. In-service inspection requirements of the ASME Code, Section XI, should be considered in the design of safety class cooling water systems.

9.1.4.4 Interfaces With Other Systems

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

- A. Reactivity Control Systems.
- B. Secondary Heat Transport System.
- C. Residual Heat Removal System.
- D. Reactor Building.
- E. ESF Ventilation System.
- F. Protection System.
- G. Non-Safety I&C System.
- H. Class IE Electric Power System.
- I. Non-Class IE Electric Power System.
- J. Spent Fuel-Handling and Storage System.
- K. Auxiliary Sodium and Purification Systems.
- L. Experimental Facilities.

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, the intent of the following GDCs that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for the design of cooling water systems:

- A. Safety Class Components.
 - 1. GDC 44, "Cooling Water"
 - 2. GDC 45, "Inspection of Cooling Water System"
 - 3. GDC 46, "Testing of Cooling Water System"
 - 4. GDC 57, "Closed System Isolation Valves"

B. Non-Safety Class Components.

1. GDC 57, "Closed System Isolation Valves"

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. Safety Class Components.

a. ASME Boiler and Pressure Vessel Code, Section X, "Fiberglass Reinforced Plastic Pressure Vessels"

2. Non-Safety Class Components.

a. ANSI B96.1-1973, "Specification for Welded Aluminum-Alloy Field Erected Storage Tanks"

b. API 620, "Recommended Rules for Design and Construction of Large, Welded Low-Pressure Storage Tanks," American Petroleum Institute, 1973

c. API 650, "Welded Steel for Oil Storage," American Petroleum Institute, 1973

d. AWWA D100-73, "Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage," American Water Works Association

B. RDT Standards.

None.

C. Guides.

1. Safety Class Components.

a. USNRC Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"

b. USNRC Regulatory Guide 1.72, "Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin"

2. Non-Safety Class Components.

None.

9.1.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 9.2.1, "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. USNRC Standard Review Plan 10.4.5, "Circulating Water System," *ibid.*
5. Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid System Outside Containment," (attached to SRP 3.6.1), *ibid.*
6. Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid Systems Piping Outside Containment," (attached to SRP 3.6.2), *ibid.*
7. IAEA Safety Guide SG-D6, Draft, "Ultimate Heat Sink," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes.

1. ANSI/ANS 59.1 (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 safety function of the fuel-handling and storage system is to handle and store 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 workers.

9.2.3 Definition

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

- A. Receiving fuel from a carrier and making necessary accountability inspections.
- B. New fuel storage prior to installation in the reactor.
- C. Handling fuel during movement between storage areas and the reactor, including core loading and unloading.
- D. In-vessel and ex-vessel spent fuel storage for irradiated fuel assemblies and blanket assemblies and other radioactive components (e.g., control assemblies and removable radial shield assemblies).

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, (2) they are required for adequate spent fuel decay heat removal, or (3) they are required for maintaining new or spent fuel in a subcritical configuration.
- B. Portions of the fuel-handling and storage system that may be located within or penetrate the reactor vessel should be designed to a safety class that is compatible with the requirements for the primary heat transport system and the reactor core assembly.
- 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.

Figure 9.2-1. Guide to the Use of Section 9.2.

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

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

A. 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).

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 to Section 4.1). The damaged cladding may be the result of a small portion of fuel element cladding that failed 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) should be limited such that the dose guidelines for the appropriate facility operating condition are not exceeded.

C. 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.

D. 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.

E. 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.

F. Interim spent fuel storage should be provided in the reactor vessel to allow for the decay of the short lived decay products. This will significantly ease the design requirements on the ex-vessel spent fuel storage facility.

G. Fuel storage facilities located within the reactor vessel should be designed such that potential interactions between the stored fuel and the reactor core are negligible.

H. The in-vessel fuel-handling machine should be suitably designed and should include suitable interlocks to preclude damaging the reactor core, fuel assemblies, reactivity control assemblies, or other safety-related equipment within the reactor vessel during fuel-handling operations.

I. Fuel-handling equipment should be designed to prevent dropping a fuel assembly during handling.

J. 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 or safety class systems 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).

K. The ex-vessel spent fuel storage facility cooling system should be designed with suitable redundancy to ensure that adequate spent fuel decay heat removal capability exists with the addition of a single failure of a component in the cooling system. Application of the single-failure criterion to system design is discussed in Chapter 15.

L. A guard vessel should be provided around the ex-vessel fuel storage facility to ensure the availability of an adequate sodium inventory for spent fuel decay heat removal in the event of a break in the ex-vessel sodium boundary.

M. The ex-vessel spent fuel storage facility should be provided with a cover gas cleanup and sodium purification system to remove radionuclides from the spent fuel cover gas and to maintain the desired coolant chemistry (see Section 9.3).

N. Provisions should be made for preheating new fuel assemblies (e.g., in an inert gas environment) prior to introducing them to a hot (e.g., 400 F) sodium environment in the ex-vessel fuel storage facility.

O. The fuel-handling and storage system should be protected against the effects of severe natural phenomena, missiles, and pipe whip, as described in Chapter 3.

P. The fuel-handling and storage system should be designed as a Seismic Category I system.

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

R. The fuel-handling and storage system should be adequately shielded to ensure that occupational exposures are maintained As Low As Reasonably Achievable (ALARA) (see Chapter 12).

9.2.4.3 Maintenance, Testing, and Inspection

Safety class components and equipment should be designed to facilitate periodic testing and inservice inspection as required by the ASME Code, Section XI. This includes the fuel storage racks, and any safety class cooling system (see Section 9.1).

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. Primary Heat Transport System.
- D. Reactor Building.
- E. ESF Ventilation System.
- F. Protection System.
- G. Non-Safety I&C System.
- H. Non-Class IE Electric Power System.
- I. Class IE Electric Power System.
- J. Cooling Water System.
- K. Auxiliary Sodium and Purification Systems.
- L. Fire Protection System.
- M. Radioactive Waste Systems.
- N. 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, the intent of the following GDC that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for the design of fuel-handling and storage components:

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

9.2.6 Standards, Codes, and Guides

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

A. Industry Codes and Standards.

1. ANSI B30.11-1973, "Safety Standard for Monorail Systems and Underhung Cranes"
2. ANSI B30.2.0-1976, "Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)"
3. ANSI B30.16-1973, "Safety Standard for Overhead Hoists"
4. ANSI N16.1-1975 (ANS 8.1), "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors"
5. ANSI N16.5-1975 (ANS 8.7), "Guide for Nuclear Criticality Safety in the Storage of Fissile Material"
6. ANSI N16.9-1975 (ANS 8.11), "Validation of Calculational Methods for Nuclear Criticality Safety"
7. ANSI N214 (ANS 54.1), Draft, "General Safety Design Criteria for an LMFBR Nuclear Power Plant"
8. CMAA Specification No. 70, "Specification for Electric Overhead Traveling Cranes," Crane Manufacturers Association of America

B. RDT Standards.

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

1. E6-24T, "Interim Decay Storage for Sodium Cooled Reactors"
2. F8-6T, "Hoisting and Rigging of Critical Components and Related Equipment"

C. Guides.

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

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

9.2.7 Supplementary Information

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

A. Background Information.

1. Branch Technical Position APCS 9-1, "Overhead Handling Systems for Nuclear Power Plants," (attached to SRP 9.1.4), NUREG 75/087, September 1975.

2. IAEA Safety Standard "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes.

1. ANSI 54.2, Draft, "Design Bases for Ex-Vessel Storage Facilities for LMFBR Spent Fuel"

9.2.8 Topics for Future Standards Development

None.

9.3 AUXILIARY SODIUM AND PURIFICATION SYSTEMS

9.3.1 Scope

This section provides guidelines for the design of auxiliary sodium and purification systems that maintain the reactor coolant chemical composition and the activity levels of cover gas systems within acceptable levels. In addition, guidelines are provided for the design of auxiliary sodium systems that maintain the desired coolant inventory in the primary heat transport system. Figure 9.3-1 is a guide to the use of this section.

9.3.2 Safety Function

The auxiliary sodium and purification systems may be required to:

A. Mitigate changes in reactor vessel sodium level (e.g., due to thermal expansion) to prevent damage to the reactor vessel thermal liner or to other components in the reactor vessel, and/or

B. Maintain activity levels in the primary sodium system cover gas within acceptable levels to ensure that the dose guidelines in Chapter 15 are not exceeded during any facility operating condition.

Figure 9.3-1. Guide to the Use of Section 9.3.

9.3.3 Definition

The Auxiliary Sodium and Purification Systems (ASPS) generally include a variety of systems such as:

- A. Sodium and NaK Receiving Systems.
- B. Primary Sodium Storage and Processing Systems.
- C. Primary Cold Trap Cooling Systems.
- D. Intermediate Liquid Metal Processing Systems.
- E. Coolant Cover Gas Processing System.

9.3.4 Design Guidelines

9.3.4.1 Guidelines for Determining the Minimum Safety Class of Auxiliary Sodium and Purification Systems (ASPS)

Because of the wide variety of components and subsystems that may be included in the set of systems referred to in the definition of the ASPS, it is not practical to specify the minimum safety class guidelines for all ASPS equipment. It is therefore recommended that the safety class be determined on a component or subsystem basis, considering the guidelines given below and the general guidelines in Table 3.2-1.

A. Portions of the auxiliary sodium and purification systems that are connected to the primary heat transport system or the spent fuel storage facility should meet the appropriate safety class requirements in Sections 5.1 and 9.2 respectively.

B. Portions of the auxiliary sodium and purification and makeup systems passing through safety class structures (e.g., the reactor building) should be designed to a safety class that is compatible with these structures.

C. Portions of the auxiliary sodium and purification systems should be designed to a minimum of Safety Class 3 if their failure would result in off-site doses in excess of 0.5 rem.

D. Other portions of the auxiliary sodium and purification systems may be non-nuclear.

9.3.4.2 Guidelines for the Design of Auxiliary Sodium and Purification Systems (ASPS)

A. The ASPS should maintain primary and intermediate liquid metal purity at desired levels. Cold traps should be used to remove impurities such as sodium oxide (Na_2O) and sodium hydride (NaH).

B. The effects of loss of cooling to cold traps (e.g., dissolution of Na_2O and NaH) should be reviewed to ensure that adequate provisions are made for cold trap isolation or restoration of cold trap cooling.

C. If necessary for protecting a reactor vessel thermal liner or other components in the reactor vessel, the ASPS should provide means for adding or removing sodium from the primary sodium system in order to maintain the desired sodium level in the reactor vessel during thermal transients (e.g., startup, shutdown, power changes).

D. Consequences of in-containment and ex-containment sodium storage tank ruptures should be reviewed to ensure adequate sodium fire protection subsystem capability (see Section 9.4).

E. The ASPS should process primary sodium cover gas to maintain desired purity and activity level. Radiation detectors should be provided to monitor the cover gas to detect signs of fuel failure.

F. The ASPS should be designed to Seismic Category I standards.

G. The ASPS should be protected against the effects of missiles and pipe whip as described in Sections 3.5 and 3.6 respectively.

H. The design of mechanical portions of the ASPS should consider the guidelines in Section 3.9.

I. Portions of the ASPS may have the capability for reactor residual heat removal. These portions should consider the guidelines in Section 5.3 for RHR systems.

J. In general, the ASPS does not require Class IE power.

9.3.4.3 Maintenance, Testing, and Inspection

The auxiliary sodium and purification systems should be designed to permit periodic tests and inservice inspections of safety class components as required by the ASME Code, Section XI.

9.3.4.4 Interfaces With Other Systems

The auxiliary sodium and purification systems may interface with:

- A. Primary Heat Transport System.
- B. Residual Heat Removal System.
- C. Non-Safety Instrumentation and Control System.
- D. Non-Class IE Electric Power System.
- E. Fuel-Handling and Storage System.

F. Radioactive Waste System.

The design of the auxiliary sodium and purification systems should assure a compatible application of codes and standards across all interfaces with other facility systems.

9.3.5 Applicable General Design Criteria

None.

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.

None.

B. RDT Standards.

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

1. A1-5T, "Purity Requirements for Operating Sodium Reactor Systems"
2. E8-13T, "Oxygen-Hydrogen Meter Module for Service in Liquid Sodium"
3. E11-2T, "Filter for Sodium Service"
4. F3-40T, "Methods for the Analysis of Sodium and Cover Gas"

C. Guides.

None.

9.3.7 Supplementary Information

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

9.4.3 Definitions

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

9.4.4 Design Guidelines

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

A. Portions of the fire protection system should be designed to a minimum of Safety Class 3 if their failure may prevent safe reactor shutdown and cooling 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 non-nuclear.

9.4.4.2 Guidelines for the Design of a Fire Protection System

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

1. Prevention of fires and combustion explosions (includes fire prevention procedures, facility design 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 fire that are not promptly detected and extinguished (includes confinement and separation features, inert gas systems, and fire resistant designs).

B. The fire protection system should include the following two subsystems:

1. Sodium Fire Protection Subsystem (SFPS).

This subsystem is designed to contain any sodium fire and minimize airborne contaminants. The SFPS may employ the following techniques for minimizing the risk for fires in liquid metal containing systems:

a. Steel cell liners and collection pans.

Liquid metal equipment cells should be steel lined if necessary to prevent hydrogen producing chemical reactions between the liquid metal and the hydrated water in the concrete cell structure. A cell may be completely lined or it may be partially lined with a steel collection pan that is sized and located to collect all potential spills into the cell.

b. Cell inerting.

Enclosed equipment cells should be provided with an inert atmosphere to prevent a self-sustaining fire. The cell inerting system may include provisions for circulating the inert gas. This feature may aid in cooling spilled sodium to the point where ignition will no longer occur.

c. Other suitable techniques for protection against sodium fires.

2. Non-Sodium Fire Protection Subsystem (NSFPS).

This subsystem is used in non-sodium containing areas of the reactor facility. The NSFPS is a conventional fire protection system.

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

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

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

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

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

1. Great care is exercised in fighting fires in areas containing liquid metal systems in order to ensure that potentially reactive fire fighting material does not come in contact with the liquid metal.

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

3. The use of combustible materials is controlled.

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

5. Proper housekeeping requirements are followed that minimize accumulation of unwanted combustibles.

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. ESF Ventilation System.
- C. Non-Safety I&C System.
- D. Non-Class IE Electric Power System.
- E. Cooling Water System.
- F. Fuel-Handling and Storage System.
- G. Radioactive Waste System.

The design of the fire protection system should assure a system 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, the intent of the following GDC that was developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for 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 A54.4-1974 (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 170-1973), "Dry Chemical Extinguishing Systems"
- k. ANSI Z285.1-1974 (NFPA 69-1973), "Explosion Prevention Systems"
- l. ANSI Z286.1-1974 (NFPA 69-1973), "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 Agency Systems"

- o. NFPA 13A-1971, "Maintenance of Sprinkler Systems"
- p. NFPA 20-1974, "Standard for the Installation of Centrifugal
Fire Pumps"
- q. NFPA 22-1974, "Water Tanks"
- r. NFPA 70B-1975, "Electrical Equipment Maintenance"
- s. NFPA 72D-1975, "Standard for the Installation Maintenance and
Use of Proprietary Protective Signaling Systems"
- t. NFPA 72E-1974, "Automatic Fire Detectors"
- u. NFPA 75-1972, "Electronic Computer/Data Processing Equipment"
- v. NFPA 80-1975, "Fire Doors and Windows"
- w. NFPA 90A-1975, "Air Conditioning and Ventilating Systems"
- x. NFPA 214-1971, "Water Cooling Towers"
- y. IEEE 634-1978, "Qualification Test for Cable Penetration Fire
Stops for Use in Nuclear Power Plants"

2. Fire Protection Administration.

- a. ANSI N45.2.3-1975, "Housekeeping During the Construction Phase
of Nuclear Power Plants"
- b. NFPA 6-1974, "Industrial Fire Loss Prevention"
- c. NFPA 6-1974, "Industrial Fire Loss Prevention"
- d. NFPA 8-1974, "Effects of Fire on Operations Management
Responsibility"
- e. NFPA 24-1973, "Standard for Outside Protection"
- f. NFPA 26-1958, "Supervision of Valves"
- g. NFPA 27-1975, "Private Fire Brigade"
- h. NFPA 30-1973, "Flammable and Combustible Liquids Code"
- i. NFPA 51B-1971, "Cutting and Welding Processes"
- j. NFPA 241-1975, "Safeguarding Buildings During Construction and
Demolition Operations"

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

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

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

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

o. NFPA 803-1978, "Fire Protection for Nuclear Power Plants"

B. RDT Standards.

None.

C. Guides.

1. Fire Protection System Equipment.

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

2. Fire Protection System Administration.

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

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

9.4.7 Supplementary Information

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

A. Background Information.

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

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

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

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

5. WASH 1243-1, "Standard for Fire Protection of AEC Electric Computer/Data Processing Systems"

B. Pending Industry Standards and Codes.

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

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

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

10.2 SAFETY FUNCTION

The safety functions of the experimental facilities include those associated with potential interfaces with the reactor core assembly (see Section 4.1), reactivity control system (see Section 4.2), primary heat transport system (see Section 5.1), the reactor building (see Section 6.2), or other safety class systems.

Test loops which form a separate "system" by themselves have the additional safety function to form a barrier against the uncontrolled release of radioactivity from the test loop such that the consequence guidelines in Chapter 15 are not exceeded for all facility operating conditions.

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, in-core circulating loops for conducting fuel tests, and ex-core auxiliary experiment or test equipment.

10.4 DESIGN GUIDELINES

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

A. In-core experimental facility components (e.g., in-core irradiation facilities and test loops) should be designed to a safety class that is compatible

Figure 10.1-1. Guide to the Use of Chapter 10.

with the safety requirements of the reactor core and the safety class of the primary heat transport system.

B. If the failure of an in-core fuel test loop or other experimental facility does not prevent safe reactor shutdown or cooling, 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.

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

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

C. The operation or failure of test or experimental equipment should not result in consequences in excess of the guidelines in Chapter 15.

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

1. Streaming from test loop components.
2. Activated portions of experimental facilities.
3. Test loop cooling water.
4. Other sources.

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

E. The design of 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 the interlock circuits interface with the protection system, the guidelines in Section 7.1 should be considered in their design.

G. Strict review procedures for experiments and experiment plans should be established to ensure that proposed experiments satisfy the following general guidelines:

1. The proposed experiments should fall within the 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. Experimental facility components that form part of the reactor core assembly or primary heat transfer system boundary should meet the maintenance, testing, and inspection requirements specified in Sections 4.1 and 5.1, respectively.

B. Prior to loading in the reactor core, experiments should be qualified for the expected environment by test, analysis, operating experience, or a combination of these methods. Testing of actual equipment under conditions which simulate the expected service conditions, to the extent practicable, (with the exception of the radiation environment), is the preferred method.

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

10.4.4 Interfaces With Other Systems

Experimental facilities may interface with the following other facility systems:

- A. Reactor Core Assembly.

- B. Reactivity Control System.

- C. Primary Heat Transport System.

- D. Reactor Building.

- E. Protection System.

- F. Non-safety I&C System.
- G. Non-safety IE Electric Power System.
- H. Cooling Water System.
- I. Radioactive Waste System.
- J. 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 GDC that are applicable to interfacing facility systems should be considered in the design of experimental facilities that may potentially affect the safe operation of the facility. In addition, the following provision of 10 CFR should be considered:

- A. 10 CFR 50.59, "Changes, Tests, and Experiments"

10.6 STANDARDS, CODES, AND GUIDES

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

- A. Industry Standards and Codes.

- 1. ANSI N16.1-1975 (ANS 8.1), "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors"

- 2. ANSI N399-1974 (ANS 15.3), "Records and Reports for Research Reactors"

- 3. ANSI N401-1974 (ANS 15.6), "Review of Experiments for Research Reactors"

- B. RDT Standards.

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

- 1. E8-11T, "FFTF Closed Loop In-Reactor Assembly Fabrication"

C. Guides.

1. USNRC Regulatory Guide 2.2, "Developments of Technical Specifications for Experiments in Research Reactors"
2. USNRC Regulatory Guide 2.4, "Review of Experiments for Research Reactors"
3. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations With Fissionable Materials Outside Reactors"

10.7 SUPPLEMENTARY INFORMATION

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

A. Background Information.

1. "Irradiation Experimentation in Fast Reactors," Proceedings of ANS Topical Meeting, September 10th to 12th, 1973, Jackson Lake Lodge, Wyoming

B. Pending Industry Standards and Codes.

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

10.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None.

CHAPTER 11

RADIOACTIVE WASTE SYSTEMS

11.1 SCOPE

The ERDA Manual, Chapter 0511, "Radioactive Waste Management," provides guidelines for establishing a program to manage radioactive waste in such a manner as to minimize the potential radiation exposure and associated risk to man. Chapter 0513 of the ERDA Manual, "Effluent and Environmental Monitoring and Reporting," prescribes monitoring requirements that are applicable to systems used to monitor radioactive waste areas. Guidelines for exposure to radiation and for maximum concentrations of radioactive material in air and water effluents are included in the ERDA Manual, Chapter 0524. These guidelines establish functional requirements for radioactive waste system designs. Radioactive waste material to be transported from a facility should be prepared for shipment in accordance with the provisions of the ERDA Manual, Chapter 0529, "Safety Standards for the Packaging of Fissile Material and Other Radioactive Material."

This chapter provides guidelines for implementing a radioactive waste management program in the design phase of 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) and As Low As Reasonably Achievable (ALARA).

11.3 DEFINITION

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

The radioactive waste system also includes monitoring and control equipment necessary for ensuring that effluents released from the radioactive waste system will not result in doses or concentrations in excess of the guidelines for the applicable facility operating condition (see Chapters 12 and 15) and are As Low As Reasonably Achievable (ALARA).

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.

B. Portions of the radioactive waste 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 radioactive waste system may be non-nuclear.

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. Primary heat transport system cover gas.
2. Liquid metal purification system.
3. Inert gas processing system.
4. ESF ventilation system and containment cleanup system (filters).
5. Fuel-handling and storage system.
6. Evaporator residues.
7. Contaminated components, or equipment used for maintenance, repair, sampling, or testing.
8. Cleaning solvents and rags.
9. Reactor building sumps.
10. Waste from radiochemical analysis.
11. 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. Portions of the radioactive waste system connected to the reactor coolant system boundary should include suitable isolation valves if required to ensure the integrity of a safety class reactor coolant system (see Section 5.1).

D. Radioactive waste storage tanks should be designed such that the failure of any one tank, containing the maximum amount of radioactive material, will not result in exposures to radioactivity in excess of the dose guidelines for the applicable facility operating condition (see Chapter 15).

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

F. Safety class radioactive waste system components and structures should be protected against the effects of severe natural phenomena and missiles (see Chapter 3).

G. Safety class radioactive waste system components and structures should be Seismic Category I.

H. The capability for concentrated liquid waste solidification and solid waste volume reduction should be provided where practical.

I. The solid radioactive waste system should include provisions for separating sodium-bearing solids from non-sodium bearing solids.

11.4.3 Maintenance, Testing, and Inspection

A. Safety class portions of the radioactive waste system should be designed to facilitate inservice inspection as required in the ASME Boiler and Pressure Vessel Code, Section XI.

B. Monitoring and control equipment should include provisions for periodic testing, inspection, and calibration.

11.4.4 Interfaces With Other Systems

The radioactive waste system may interface with the following 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. Auxiliary Sodium and Purification Systems.
- H. Experimental Facilities.
- I. 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, the intent of the following GDC and other provisions of 10 CFR that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for the design of radioactive waste systems:

- A. GDC 60, "Control of Releases of Radioactive Materials to the Environment"
- B. GDC 61, "Fuel Storage and Handling and Radioactivity Control"
- C. GDC 63, "Monitoring Fuel and Waste Storage"
- D. GDC 64, "Monitoring Radioactivity Releases"
- E. 10 CFR 20, "Standards for Protection Against Radiation"
- F. 10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Materials in Effluents - Nuclear Power Plants"
- G. 10 CFR 50.36a, "Technical Specifications on Effluents from Nuclear Power Plants"
- H. 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Reasonably Achievable' for Radioactive Material in Light Water-Cooled Nuclear Power Reactor Effluents"

11.6 STANDARDS, CODES, AND GUIDES

In addition to the standards, codes, and guides listed in Chapter 3, it is recommended that the following standards, codes, and guides be considered in the design of radioactive waste systems:

- A. Industry Standards and Codes.
 - 1. ANSI N5.8-1967, "Definition of Radioactive Waste Categories"

B. RDT Standards.

None.

C. Guides.

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

2. USNRC Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Consequences of a PWR Radioactive Gas Storage Tank Failure"

3. USNRC Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light Water-Cooled Nuclear Power Reactors"

4. USNRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures and Components Installed in Light Water-Cooled Nuclear Power Plants"

11.7 Supplementary Information

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

A. Background Information.

1. USNRC Standard Review Plan 11.2, "Liquid Waste Management Systems," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 11.3, "Gaseous Waste Management Systems," *ibid.*

3. USNRC Standard Review Plan 11.4, "Solid Waste Management Systems," *ibid.*

4. USNRC Standard Review Plan 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems," *ibid.*

5. WASH-1202, "Plan for the Management of AEC-Generated Radioactive Wastes"

6. Moghissi, A.A., Godbee, H.W., Ozker, M.S., Carter, M.W., et. al., "Nuclear Power Waste Technology," American Society of Mechanical Engineers, New York, 1978

B. Pending Industry Standards and Codes.

1. ANS N498 (ANS 40.12), Proposed, "General Criteria for Waste Management"
2. ANSI N525 (ANS 40.11), Proposed, "Definition of Radioactive Waste Categories"

11.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None.

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CHAPTER 12

RADIATION PROTECTION

12.1 SCOPE

Requirements for radiation protection are specified in the following ERDA Manual chapters:

- A. 0502, "Notification, Investigation and Reporting of Occurrences"
- B. 0513, "Effluent and Environmental Monitoring and Reporting"
- C. 0524, "Standards for Radiation Protection"
- D. 0525, "Occupational Radiation Exposure Information"
- E. 0545, "Nuclear Accident Dosimetry Program"

This chapter provides additional design guidelines for radiation protection during normal facility operations, and includes references to detailed design standards that are applicable to radiation protection systems and procedures. Radiological consequence guidelines during postulated accidents are included in Chapter 15. Figure 12.1-1 is a guide to the use of this chapter.

12.2 SAFETY FUNCTION

The safety function of a radiation protection system is to minimize the exposure of man to radiation, considering all potential sources of radiation and radioactive material.

12.3 DEFINITION

The hardware portion of a radiation protection system includes the components and systems required for monitoring, surveying, sampling, and testing the radiation environment of a facility and its effluents. The system generally includes provisions for alarming abnormal conditions. Portions of the radiation protection system may be part of the protection system (see Section 7.1) and may automatically initiate protective actions. The radiation protection system hardware also includes permanent and temporary shielding installed to minimize personnel exposure to direct and scattered radiation.

The administrative portion of the radiation protection system includes a system of records, policies and operating procedures which facilitate maintaining occupational and off-site exposures to radiation and radioactive material As Low As Reasonably Achievable (ALARA).

Figure 12.1-1. Guide to the Use of Chapter 12.

12.4 DESIGN GUIDELINES

12.4.1 Guidelines for the Design of a Radiation Protection System

A. The radiation protection system, incorporating administrative procedures and protective hardware, should be designed to assure that the exposure of personnel to radiation is maintained within the following guidelines:

1. Occupational Radiation Exposure Guidelines.

Exposure of personnel in controlled areas should be maintained ALARA in accordance with the ERDA Manual, Chapter 0524, Part V and USNRC Regulatory Guide 8.8.

2. Off-Site Radiation Exposure Guidelines.

Exposure of persons in unrestricted areas to routine releases of radioactive material should be maintained ALARA in accordance with the ERDA Manual, Chapter 0524, Part V and 10 CFR 50, Appendix I.

B. The facility design should ensure adequate protection from direct and scattered radiation and airborne sources. Methods of protection to be considered in facility design include the following:

1. Permanent or temporary shielding.
2. Ventilation systems (see Section 6.3).
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 (may actuate a containment isolation system and/or an ESF ventilation system).
2. Intermediate heat transport loop activity level (may result from primary sodium leakage into intermediate cooling loop).
3. Radioactive waste system areas.

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

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

The design of the radiation protection system should ensure a compatible application of codes and standards across 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, the intent of the following provisions of 10 CFR that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific GDC for the design of a radiation protection system:

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

- B. GDC 63, "Monitoring Fuel and Waste Storage"
- C. GDC 64, "Monitoring Radioactivity Releases"
- D. 10 CFR 19, "Notices, Instructions and Reports to Workers; Inspections"
- E. 10 CFR 20, "Standards for Protection Against Radiation"
- F. 10 CFR 30, "Rules of General Applicability to Licensing of Byproduct Material"
- G. 10 CFR 40, "Licensing of Source Material"
- H. 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Reasonably Achievable' for Radioactive Material in Light Water-Cooled Nuclear Power Reactor Effluents"
- I. 10 CFR 70, "Special Nuclear Material"
- J. 10 CFR 100, "Reactor Site Criteria"

12.6 STANDARDS, CODES, AND GUIDES

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

- A. Industry Standards and Codes.
 - 1. Radiation Protection System Equipment.
 - a. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Material of Nuclear Facilities"
 - b. ANSI N13.3-1969, "Dosimetry for Criticality Accidents"
 - c. ANSI N13.4-1971, "Specification for Portable X- and Gamma-Radiation Survey Instruments"
 - d. ANSI N13.5-1972, "Performance Specification for Direct Reading and Indirect 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 (ANS 8.5), "Criticality Accident Alarm System"

- h. ANSI N101.6-1972, "Concrete Radiation Shields"
 - i. ANSI N237-1975 (ANS 18.1), "Source Term Specification"
 - j. ANSI N319-1976, "Personnel Neutron Dosimeters"
 - k. ANSI N322-1977, "Inspection and Test Specifications for Direct and Indirect Reading Quartz Fiber Pocket Dosimeters"
 - l. ANSI N512-1974, "Protective Coatings (Paints) for the Nuclear Industry"
 - m. ANSI N648-1975, "Specification for Aggregates for Radiation Shielding Concrete"
 - n. ANSI N649-1975, "Descriptive Nomenclature of Constituents of Aggregates for Radiation Shielding Concrete"
 - o. ANSI/ANS 6.4-1977 (N403), "Guidelines on 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 (ANS 3.1), "Selection and Training of Personnel for Nuclear Power Plants"
 - g. ANSI N18.9-1972 (ANS 6.3), "Program for Testing Biological Shielding in Nuclear Power Plants"
 - h. ANSI N105-1973, "Definition of Terms Relating to Dosimetry"
 - i. ANSI N323-1978, "Radiation Protection Instrumentation Test and Calibration"
 - j. ANSI N343-1978, "Internal Dosimetry for Mixed Fission and Activation Products"
 - k. ANSI N510-1975, "Testing Nuclear Air Cleaning Systems"

1. ANSI Z88.2-1969, "Procedures for Respiratory Protection"
- m. ANSI/ANS 6.1.1-1977 (N666), "Neutron and Gamma-Ray Flux-to-Dose Rate Factors"

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 System"

- 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 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 3.2, "Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles"
- i. USNRC Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants"
- j. USNRC Regulatory Guide 4.8, "Environmental Technical Specifications for Nuclear Power Plants"
- k. USNRC Regulatory Guide 8.1, "Radiation Symbol"
- l. USNRC Regulatory Guide 8.2, "Guide for Administrative Practices in Radiation Monitoring"
- m. USNRC Regulatory Guide 8.5, "Immediate Evacuation Signal"
- n. USNRC Regulatory Guide 8.7, "Occupational Radiation Exposure Records System"
- o. USNRC Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable"
- p. USNRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable"
- q. USNRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection"
- r. USNRC Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Plants Design Stage Man-Rem Estimates"

12.7 SUPPLEMENTARY INFORMATION

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

A. Background Information.

1. USNRC Standard Review Plan 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems," NUREG 75/087, September 1975
2. USNRC Standard Review Plan 12.1, "Assuring that Occupational Radiation Exposures are ALARA," *ibid.*
3. USNRC Standard Review Plan 12.2, "Radiation Sources," *ibid.*
4. USNRC Standard Review Plan 12.3, "Radiation Protection Design Features," *ibid.*
5. USNRC Standard Review Plan 12.4, "Dose Assessment" *ibid.*
6. USNRC Standard Review Plan 12.5, "Health Physics Program," *ibid.*
7. Branch Technical Position ETSB 11-2, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Reactor Plants," (attached to SRP 11.3), *ibid.*
8. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, March 23, 1962
9. Schaffer, N.M., ed., "Reactor Shielding for Nuclear Engineers," USAEC, 1973
10. "Respiratory Protective Services for Use in Atmospheres Containing Radioactive Materials," USBM-23, U.S. Bureau of Mines, 1973
11. UL-586, "High Efficiency Particulate, Air Filter Units," Underwriter's Laboratories, Inc.
12. UL-900, "Air Filter Units," Underwriter's Laboratories, Inc.
13. Burchsted, C.A., et al., "Nuclear Air Cleaning Handbook," ERDA 76-21, Oak Ridge National Laboratory, March 31, 1976
14. WASH-1287, "Manual of Respiratory Protection Against Airborne Radioactive Material," USAEC, 1974
15. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

16. IAEA Safety Guide SG-05, Draft, "Operational Aspects of Radiological Protection"

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 N13.15, Draft, "Performance of Personnel Thermoluminescence Dosimetry Systems"

2. ANSI N320, Draft, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation"

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

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

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

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

9. ANSI/ANS 6.8.1, Draft, "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors"

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

12.8 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None.

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CHAPTER 13

CONDUCT OF OPERATIONS

13.1 SCOPE

The following ERDA Manual chapters provide guidance for the conduct of facility operations:

- A. 0502, "Notification, Investigation, and Reporting of Occurrences"
- B. 0530, "Nuclear Criticality Safety"
- C. 0531, "Safety of Nonreactor Nuclear Facilities"
- D. 0540, "Safety of AEC-Owned Reactors"
- E. 0550, "Operational Safety Standards"
- F. 0601, "Emergency Preparedness Programs"
- G. Immediate Action Directives 8401-6 and 8401-7

This chapter provides a listing of other standards and guides that are recommended for consideration in the conduct of facility operations.

13.2 APPLICABLE GENERAL CRITERIA

It is recommended that the intent of the following provisions of 10 CFR that were developed specifically for water-cooled reactors be considered by the reactor staff in the development of specific requirements for 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, "Operator's License"
- G. 10 CFR 73, "Physical Protection of Plants and Materials"

13.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in the conduct of facility operations:

A. Industry Standards and Codes.

1. ANSI N13.2-1969, "Administrative Practices in Radiation Monitoring"
2. ANSI N15.8-1974, "Nuclear Material Control Systems for Nuclear Power Plant"
3. ANSI N15.26-1976, "Physical Protection of Special Nuclear Material Within a Facility"
4. ANSI N16.1-1969 (ANS 8.1), "Nuclear Criticality Safety Standard for Operations With Fissionable Materials Outside Reactors"
5. ANSI N18.7-1976 (ANS 3.2), "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"
6. ANSI N18.17-1973 (ANS 3.3), "Industrial Security for Nuclear Power Plants"
7. ANSI N399-1974 (ANS 15.3), "Records and Reports for Research Reactors"
8. ANSI N401-1974 (ANS 15.6), "Review of Experiments for Research Reactors"
9. ANSI N524-1976 (ANS 18.2), "Nuclear Plant Reliability Data Collection and Reporting System"
10. ANSI N546-1976 (ANS 3.4) "Medical Certification and Monitoring for Personnel Requiring Operator Licenses for Nuclear Power Plants"
11. ANSI/ANS 3.1-1978, "Selection and Training of Nuclear Power Plant Personnel"
12. ANSI/ANS 3.5-1979, "Nuclear Power Plant Simulators for Use in Operator Training"

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. F3-20T, "Operation and Maintenance Manuals"

4. F5-9T, "Sodium Removal Process"

C. Guides.

1. USNRC Regulatory Guide 1.8, "Personnel Selection and Training"

2. USNRC Regulatory Guide 1.16, "Reporting of Operating Information"

3. USNRC Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage"

4. USNRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)"

5. USNRC Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants"

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

7. USNRC Regulatory Guide 1.114, "Guidance for Being Operator at the Controls of a Nuclear Power Plant"

8. USNRC Regulatory Guide 1.134, "Medical Certification and Monitoring of Personnel Requiring Operator Licenses"

9. USNRC Regulatory Guide 2.4, "Review of Experiments for Research Reactors"

10. USNRC Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations With Fissionable Material Outside Reactors"

11. USNRC Regulatory Guide 5.7, "Control of Personal Access to Protected Areas, Vital Areas, and Material Access Areas"

12. USNRC Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials"

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

14. USNRC Regulatory Guide 5.54, "Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants"

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

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

17. USNRC Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable"

13.4 SUPPLEMENTARY INFORMATION

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

A. Background Information.

1. "Safe Operation of Nuclear Power Plants," IAEA Safety Series No. 31, International Atomic Energy Agency, Vienna, 1969
2. "Safe Operation of Critical Assemblies and Research Reactors, 1971 Edition," IAEA Safety Series No. 35, International Atomic Energy Agency, Vienna, 1971
3. "Manual for the Operation of Research Reactors," IAEA Technical Report Series No. 37, International Atomic Energy Agency, Vienna, 1964
4. "Planning for the Handling of Radiation Accidents," IAEA Safety Series No. 32, International Atomic Energy Agency, Vienna, 1969
5. IAEA Safety Standard 50-C-0, "Safety in Nuclear Power Plant Operation, Including Commissioning and Decommissioning, A Code of Practice," International Atomic Energy Agency, Vienna
6. IAEA Safety Guide SG-01, Draft, "Staffing, Recruitment, Training, and Authorization of Operating Personnel"
7. IAEA Safety Guide SG-03, Draft, "Operational Limits and Conditions"
8. IAEA Safety Guide SG-06, Draft, "Emergency Arrangements of the Operating Organization"
9. IAEA Safety Guide SG-07, Draft, "Maintenance of Nuclear Power Plants"
10. IAEA Safety Guide SG-G6, Draft, "Emergency Preparedness of Public Authorities"
11. USNRC Standard Review Plan 13.1.1, "Management and Technical Support Organization," NUREG 75/087, September 1975
12. USNRC Standard Review Plan 13.1.2, "Operating Organization," *ibid.*
13. USNRC Standard Review Plan 13.1.3, "Qualifications of Nuclear Plant Personnel," *ibid.*

14. USNRC Standard Review Plan 13.2, "Training," ibid.
15. USNRC Standard Review Plan 13.3, "Emergency Planning" ibid.
16. USNRC Standard Review Plan 13.4, "Review and Audit" ibid.
17. USNRC Standard Review Plan 13.5, "Plant Procedures" ibid.
18. USNRC Standard Review Plan 13.6, "Industrial Security" ibid.
19. EPA 520/1-75-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents" U.S. Environment Protection Agency, September 1975
20. NUREG-0094, "A Guide to the Licensing of Facility Operators, Including Senior Operators" USNRC, July 1976
21. WASH 1130, Revised, "Utility Staffing and Training for Nuclear Power" USAEC, June 1973

B. Pending Industry Standards and Codes.

1. ANSI N514 (ANS 3.7), Proposed, "Guide to Standard Format and Content of Emergency Plans for Nuclear Power Generating Stations"
2. ANSI N682 (ANS 3.7.1), Proposed, "Facilities for Medical Care for On-Site Nuclear Power Plant Radiological Emergencies"
3. ANSI N684 (ANS 3.7.2), Proposed, "Emergency Control Center for Nuclear Power Plants"
4. ANSI N685 (ANS 3.7.3), Proposed, "Radiological Emergency Preparedness Exercises for Nuclear Power Plants"

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

The intent of the following provisions of 10 CFR that were developed specifically for water-cooled reactors should be considered by the reactor staff in the development of specific criteria for the initial test:

- 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 (ANS 3.2), "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"
2. IEEE 336-1977 (ANSI N45.2.4), "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
3. IEEE 415-1976, "Guide for Planning of Preoperational Testing Programs for Class IE Power Systems for Nuclear Power Generating Stations"
4. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components"

B. RDT Standards.

None.

C. Guides.

1. USNRC Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"
2. USNRC Regulatory Guide 1.41, "Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments"
3. USNRC Regulatory Guide 1.80, "Preoperational Testing of Instrument Air System"

14.4 SUPPLEMENTARY INFORMATION

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

A. Background Information.

1. USNRC Standard Review Plan 14.1, "Initial Plant Test Programs-PSAR" NUREG 75/087, September 1975
2. USNRC Standard Review Plan 14.2, "Initial Plant Test Program-FSAR" *ibid.*
3. 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 (e.g., a rod withdrawal accident may be accommodated within the consequence guidelines for an emergency operating condition). With the addition of a single failure (discussed in Section 15.3), the 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, or it may be necessary to design the system to a final safety class that is higher than the minimum safety class recommended by this Design Guide in order to improve system quality and reliability.

B. Anticipated Transients Without Scram (ATWS) generally need not be postulated for Category IV reactor if the recommendations of Section 4.2, regarding the use of at least two independent, diverse drivers and functionally redundant reactor shutdown systems, have been met. If the recommendations of Section 4.2 have not been met, it is recommended that the following ATWS events be analyzed:

1. Inadvertent insertion of positive reactivity by the reactivity control system.
2. Loss of intermediate coolant or secondary coolant system flow.
3. Loss of ac power.
4. Loss of electrical load (facilities with turbine generator only).

Table 15.2-1. Spectrum of Postulated Accidents to be Considered in the Design of Category IV 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 accidents in which a reactivity control assembly inadvertently drops into the reactor core. 6. Spectrum of reactivity insertion accidents due to operation or failure of experimental facilities.
B. Radioactive Release from a Subsystem or Component
<ol style="list-style-type: none"> 1. Radioactive gas waste system leak or failure. 2. Radioactive liquid waste system leak or failure. 3. Fuel handling accidents. 4. Accidents involving fuel stored in the reactor vessel. 5. Spent fuel cask drop accidents. 6. Experimental facility (or test loop) failures.

Table 15.2-1. Spectrum of Postulated Accidents to be Considered in the Design of Category IV Reactor Facilities. (Continued)

C. Decrease in Primary Sodium System Flow Rate	
1.	Single and multiple sodium pump trips or mechanical failures at power.
2.	Partial or complete loss of primary sodium pumping capability when shutdown.
3.	Reactor coolant system flow blockage due to failure of in-core experiment facility or other credible cause.
D. Decrease in Primary Sodium Inventory	
1.	Break in instrument line or other small line connected to the reactor coolant boundary.
2.	Major primary sodium pipe failure.
E. Increase in Reactor Coolant Inventory	
1.	Intermediate sodium heat exchanger tube failure.
F. Decrease in Heat Removal by the Intermediate Liquid Metal Cooling Loop	
1.	Inadvertent isolation of an intermediate loop.
2.	Partial or complete loss of intermediate loop pumping power.
3.	Partial or complete loss of intermediate loop heat sink (sodium-air heat exchanger or secondary coolant system, as appropriate)

Table 15.2-1. Spectrum of Postulated Accidents to be Considered in the Design of Category IV Reactor Facilities. (Continued)

<p>G. Decrease in Heat Removal by the Secondary (Steaming) Cooling System (if this system is provided)</p>
<ol style="list-style-type: none"> 1. Inadvertent closure of a steam generator isolation valve. 2. Steam generator tube leak or failure (secondary system-to-intermediate loop leaks). 3. Partial or complete loss of secondary cooling system pumping power. 4. Steam pressure regulator malfunction or failure that results in decreasing steam flow. 5. Loss of external electric load. 6. Turbine trip (stop valve closure). 7. Loss of condenser vacuum.
<p>H. Increase in Heat Removal by the Intermediate and Secondary Coolant System</p>
<ol style="list-style-type: none"> 1. Intermediate and/or secondary coolant system failures that result in excessive heat removal from the primary sodium system.

Table 15.2-2. Summary of Facility Operating Conditions and the Corresponding Frequency of Occurrence and Consequence Guidelines.

Operating Condition	Description	Typical Frequency of Occurrence, F (yr ⁻¹)*	Typical Design Basis Events	Radiological Consequence Guidelines	
				Off-Site	On-Site
Normal	Events that are planned to occur regularly in the course of plant operation.	$F \geq 1$	Startup, shutdown, 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	$1 > F \geq 10^{-1}$	Loss of electric power, partial loss of core cooling, inadvertent reactor shutdown	ALARA** <ERDAM 0524	ALARA** <ERDAM 0524
Emergency	Infrequent incidents or occurrences which may occur during the lifetime of a facility	$10^{-1} > F \geq 10^{-3}$	Loss of normal core cooling	Small fraction of 10CFR100	Small fraction of on-site faulted dose guidelines
Faulted	Events which are limiting faults and are not expected to occur during the life of a facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material and because they represent upper bounds on failures or accidents with a probability of occurrence sufficiently high to require consideration in design. Generally a postulated accident is not considered in design if the probability of its occurrence resulting in unacceptable consequences is less than 10^{-7} per year	$10^{-3} > F \geq 10^{-7}$	Design basis reactivity accident	<10CFR100	<25 rem whole body, <300 rem thyroid, <150 rem bone, <75 rem lung, and <75 rem all other organs

* The tabulated frequency ranges for each operating condition are illustrative examples only. They are based on the written description in the table, for a facility with an expected lifetime in the 20-40 year range.

** As low as reasonably achievable.

Table 15.2-2. (Continued). Summary of Facility Operating Conditions and the Corresponding Frequency of Occurrence and Consequence Guidelines

Operating Condition	General Guidelines	Fuel Design Guidelines	Mechanical System Design Guidelines	Containment 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 (e.g., ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsections NB, NC, and ND for Safety Class 1, 2, and 3 components, respectively - see summary of limits and loading combinations in Tables 3.9-2 and 3.9-3)	Maximum containment loadings within applicable design code allowable (e.g., ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE or Division 2 for Metal Containments and Concrete Containments, respectively, see Sections 3.8 and 6.2)
Upset	The facility should be capable of returning to operation without extensive corrective action or repair			
Emergency	Facility should be capable of returning to operation following corrective action or repair of damage	Any fuel or cladding damage should not prevent reactor shutdown and adequate core cooling (see Section 4.1)		
Faulted	Facility damage may preclude return to operation			

5. Turbine trip (facilities with turbine generator only).

C. The radiological consequences of the postulated accidents should not exceed the guidelines in Table 15.2-2. The dose guidelines for the faulted facility operating condition are discussed in detail below:

1. On-Site Dose Guidelines.

In determining the on-site dose consequences that may result from postulated design basis accidents, appropriate credit may be taken for the site emergency plan and the ability to transport and evacuate on-site personnel. Considering this ability to manage on-site personnel, it is recommended that the accident dose consequence goals be considerably less than the faulted operating condition dose guidelines in Table 15.2-2 (25 rem total radiation dose to the whole body, 300 rem total radiation dose to the thyroid from iodine exposure, 150 rem to the bone and 75 rem to the lung from plutonium exposure, and 75 rem to any other organ).

For individuals who constitute the site emergency force, the guidance in ERDAM 0524, (particularly in Part IV, "Guidance for Emergency Exposures During Rescue and Recovery Activities") should be considered in determining the dose guidelines that are applicable to emergency force members when performing their duties during an emergency in response to a postulated design basis accident.

2. Off-Site Dose Guidelines.

a. At the exclusion area boundary (see Section 2.1), an individual remaining at any point on the boundary for two hours immediately following the onset of the design basis accident will not receive a total radiation dose to the whole body in excess of 25 rem, a total radiation dose in excess of 300 rem to the thyroid from iodine exposure, 150 rem to the bone and 75 rem to the lung from plutonium exposure, and 75 rem to any other organ.

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 resulting from the postulated fission product release resulting from the design basis accident (during the entire period of its passage) will not receive a total radiation dose to the whole body in excess of 25 rem, a total radiation dose in excess of 300 rem to the thyroid from iodine exposure, a radiation dose in excess of 150 rem or 75 rem to the lung from plutonium exposure, or a radiation dose in excess of 75 rem to any other organ.

3. Dose Equivalence.

Following some postulated accidents, the dose to the critical organ due to inhalation or ingestion of radionuclides may be more limiting than the whole body dose or the thyroid dose guidelines for the appropriate operating condition. For these accidents, the critical organ should be identified and the organ dose limited such that it does not exceed a value that would be comparable to the corresponding whole body dose or thyroid dose guidelines.

D. The minimum safety class to which components, systems, and structures should be designed is determined according to the guidelines in Section 3.2. As recommended in that section, a system which contains or may contain radioactivity, may be designed to non-nuclear standards if the maximum off-site radiological consequences of system failure do not exceed 0.5 rem to the whole body, and if the system is not otherwise required to be a 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.

The final safety classification of a system is obtained by analysis of the design basis accidents. Because of specific features of an individual Category IV 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., improve containment isolation provisions) to ensure a final 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.

E. The spectrum of design basis accidents to be considered in the design of Category IV reactors is listed in Table 15.2-1. Events such as floods, storms, earthquakes, and fires are not explicitly considered in this chapter. Severe natural phenomena (floods, storms, earthquakes) and combinations of natural and external man-made hazards are discussed in Chapter 3 and fires are discussed in Section 9.4. 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 paragraph G.

On the basis of the likelihood of occurrence during the life of a reactor facility, design basis accidents may be classed as upset, emergency, or faulted operating conditions.

The descriptions of the facility operating conditions in Table 15.2-2 include very broad guidelines for determining the facility operating condition to which a particular accident should be assigned. On the basis of these broad guidelines, Table 15.2-2 additionally provides numerical examples of the range of annual frequency of occurrence that may typically be expected for accidents in each of the facility operating conditions. The actual frequency of occurrence range corresponding to each operating condition should be determined for each individual reactor facility based on the expected facility operating lifetime and the nature and consequences of accidents postulated for the facility.

This Guide identifies applicable General Design Criteria from 10 CFR 50, Appendix A. These criteria refer to Anticipated Operational Occurrences (AOO's). Appendix A of 10 CFR 50 defines AOO's as events which are expected to occur one or more times in the life of the reactor facility. Therefore, AOO's include normal, upset, and the more likely emergency events. The relationship between AOO's and these facility operating conditions is illustrated in Figure 15.2-1.

Figure 15.2-1. Approximate Relationship Between the Facility Operating Conditions and the Anticipated Operational Occurrences.

Note: These frequency ranges for each operating condition and for AOOs are illustrative examples only. They are based on the information in Table 15.2-2.

F. The spectrum of design basis accidents and severe natural phenomena considered in facility design is acceptable if it includes all postulated types of accidents for which a realistic estimate of the probability of occurrence of potential doses comparable to the faulted operating condition guidelines exceeds approximately 10^{-7} per year.

In view of the low probability events under consideration, the probability of occurrence of the initiating events leading to these potential radiological consequences should be estimated using assumptions that are as realistic as is practicable. In addition, because of these low probability events, valid statistical data are often not available to permit accurate quantitative calculations of probabilities. Accordingly, a conservative calculation showing that the probability of occurrence of potential doses comparable to the faulted operating condition guidelines is approximately 10^{-6} per year is acceptable if, when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower.

The overall safety objective for an individual facility should be to ensure that there is less than one chance in a million per year ($<10^{-6}$ per year) of a serious accident.

G. 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 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 accident analysis should be at a conservative margin above the actual maximum operating power level in order to account for instrument error and other factors.

3. The initial conditions for the accident analysis (before the initiating event) should be within the range of operating conditions permitted by the facility technical specifications.

4. Conservative time intervals for safety-related operator actions that are required to mitigate the consequences of postulated accidents should consider the guidance in ANSI N660.

15.3 GUIDELINES FOR APPLICATION OF THE SINGLE-FAILURE CRITERION TO ACCIDENT ANALYSIS AND SYSTEM DESIGN

A. General.

As defined in 10 CFR 50, Appendix A, a single failure is an occurrence resulting in the loss of capability of a component to perform its intended safety

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 require that systems such as the residual heat removal system, cooling water systems, 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.

B. Types of Single Failures.

Single failures may occur in either active or passive components. An active component is defined as one which requires physical motion of some sort to perform its intended function (e.g., motor operated valves, electrical relays, pumps). Failure of an active component means failure to provide the physical motion required for the safety function, not the failure of the passive parts (e.g., a pressure boundary) of the active component. A passive component, such as a pipe, tank, accumulator, valve body, or electrical cable does not require physical motion to perform its function.

Active single failures may include failure to act or incorrect action when required, and spurious action or operation when not required. Examples of single active failures are failure of a pump or a standby diesel generator to start in response to a valid signal and the operation of a motor-operated valve in the absence of a valid signal. Passive single failures may include ruptures or other failures of fluid system pressure boundaries, flow backage 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 procedures for conservatively determining the effects of a postulated accident and the effectiveness of safety class systems in mitigating potential consequences includes the following steps:

1. The initiating event is postulated. It may be assumed to occur in a safety class system (e.g., a reactivity transient caused by an uncontrolled rod withdrawal).
2. Other failures that are a direct consequence of the initiating event are then assumed to take place.

3. In response to the postulated accident, safety class systems may be called into operation to mitigate the accident consequences. To ensure a conservative design of these systems, a single failure is assumed to occur in a safety class system thereby diminishing its capabilities.

4. Consequences of the postulated accident must then be demonstrated to be within the corresponding guidelines specified in Table 15.2-2.

10 CFR 50, Appendix A, states that fluid and electric systems are considered to be designed against an assumed single failure if neither a single failure of any active component (assuming passive components function properly) nor a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety function.

Some single failures, such as a failure of a standby diesel generator to start, or a single failure of any other type of emergency electric power supply, may disable a protection system channel and any redundant safety system trains powered from the power supply. This multiple failure, resulting from a single occurrence, is still considered to be a "single failure."

The basic statement of the single-failure criterion is concise; however, its effects are far reaching and often require interpretation to permit reasonable implementation in the design of reactor facility systems. Additional guidance on the application of the single-failure criterion is found in USNRC Regulatory Guide 1.53 and IEEE 379.

Safety class system failures may be classified as either detectable or undetectable failures. All potential single failures are detectable failures and may be discovered by periodic tests, anomalous indications, or alarms. For the purpose of system analysis, all potential undetectable failures should be identified by a failure mode and effects analysis. 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 their 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 may be divided into two phases - the "short term" and the "long term."

The short term is typically a period of operation up to 24 hours following an accident initiating event during which safety systems must function to place

the facility in a safe condition. The long term is a period of operation following the short term during which continued operation of some safety systems is required in order to maintain the facility in a safe condition.

During the short term, the facility should be designed to accommodate the postulated accident within the applicable consequences 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, or a pump to start).

Passive fluid system single failures 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 a 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 a containment isolation system.

During the long term, assuming no prior application of the single-failure criterion during the short term, the single failure to be considered may be (1) any electrical single failure, (2) any active fluid system failure, or (3) a passive fluid system failure (e.g., a failure of a pressure boundary component such as a valve packing, a pump mechanical seal, or other leak source).

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 APPLICABLE GENERAL CRITERIA

The intent of the following provisions of 10 CFR that were developed specifically for water-cooled reactors should be considered by the reactor staff when performing or evaluating the analysis of postulated accidents:

- A. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants"
- B. 10 CFR 100, "Reactor Site Criteria"

15.5 STANDARDS, CODES, AND GUIDES

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

- A. Industry Standards and Codes.

- 1. ANSI/ANS 4.1-1978, "Design Basis Criteria for Safety Systems in Nuclear Power Generating Stations"

2. ANSI N660 (ANS 51.4), Draft, "Criteria for Safety-Related Operator Actions"

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

B. RDT Standards.

None.

C. Guides.

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

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

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

15.6 SUPPLEMENTARY INFORMATION

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

A. Background Information.

1. USNRC Standard Review Plan 2.2.3, "Evaluation of Potential Accidents," NUREG 75/087, September 1975

2. USNRC Standard Review Plan 2.3.4, "Short Term Diffusion estimates," *ibid.*

3. Branch Technical Position EICSB 18, "Application of the Single-Failure Criteria to Manually Controlled Electrically Operated Valves," (in SRP, Appendix A), *ibid.*

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

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

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

7. Sesonske, A., "Nuclear Power Plant Design Analysis," TID-26241, USAEC, 1973

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

9. Streng, D.L., et al., "Review of Calculational Models and Computer Codes for Environmental Dose Assessments of Radioactive Releases," BNWL-B-454, Battelle Northwest Laboratories, June 1976

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

11. Fish, B.R., et al., "Calculation of Doses Due to Accidentally Released Plutonium from an LMFBR," ORNL-NSIC-74, Oak Ridge National Laboratory, November 1972

12. IAEA Safety Standard 50-C-D, "Design for Safety of Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

B. Pending Industry Standards and Codes.

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

2. ANS 54.7, Draft, "Site Suitability Source Terms"

3. ANS 54.10, Proposed, "Risk Limit Guidelines for LMFBR Design"

4. ANS 54.13, Proposed, "Evaluation of Potential Radiological Consequences of LMFBR Radioactive Gas Process and Storage System Failures"

15.7 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

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

A. Guidelines for a standard approach to calculating doses from postulated Category IV reactor accidents.

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CHAPTER 16

TECHNICAL SPECIFICATIONS

16.1 SCOPE

The ERDA Manual, Chapter 0540, "Safety of AEC-Owner Reactors," requires that the limitations for safe operation be specified in a Technical Specification document for each reactor, with the detailed requirements being commensurate with the potential risks that may arise from facility operation.

This chapter provides a listing of other guides and standards that are recommended for consideration when establishing technical specification limits for safe reactor operation.

16.2 APPLICABLE GENERAL CRITERIA

The intent of the following provisions of 10 CFR that were developed specifically for water-cooled reactors should be considered by the reactor staff in establishing technical specifications for reactor facilities:

A. 10 CFR 50.36, "Technical Specifications"

B. 10 CFR 50, Appendix I, Section IV, "Guides on Technical Specifications for Limiting Conditions for Operation for Light Water-Cooled Power Reactors Licensed Under 10 CFR Part 50"

16.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in establishing technical specifications for reactor facilities:

A. Industry Standards and Codes.

1. ANSI/ANS 58.4-1979, "Criteria for Technical Specifications for Nuclear Power Stations"

B. RDT Standards.

None.

C. Guides.

1. USNRC Regulatory Guide 1.16, "Reporting of Operating Information," Appendix A, Technical Specifications

2. USNRC Regulatory Guide 1.105, "Instrument Setpoints"

3. USNRC Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors"

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

16.4 SUPPLEMENTARY INFORMATION

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

A. Background Information.

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

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

3. Branch Technical Position EICSB-6, "Capacity Test Requirements for Station Batteries, Technical Specifications," *ibid.*

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

B. Pending Industry Standards and Codes.

None.

16.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None.

CHAPTER 17

QUALITY ASSURANCE

17.1 SCOPE

The ERDA Manual, Chapter 0540, "Safety of AEC-Owned Reactors," requires that quality assurance programs be developed and implemented for reactor projects. The implementation of the Quality Assurance program is described in the ERDA Manual, Chapter 0820, "Quality Assurance."

This chapter lists other standards and guides that are recommended for consideration when establishing QA programs for safety class components, systems, and structures during the design, constructions, and operation phases of a reactor facility. Non-safety class (non-nuclear) components, systems, and structures may utilize normal industrial quality assurance procedures in lieu of the codes, standards, and guides listed in this section providing they meet the requirements of ERDAM 0820.

17.2 APPLICABLE 10 CFR GENERAL CRITERIA

The intent of the following provisions of 10 CFR that were developed specifically for water-cooled reactors should be considered by the reactor staff in establishing quality assurance programs for the design construction, and operation phases of reactor facilities:

- A. 10 CFR 50.55a, "Codes and Standards"
- B. 10 CFR 50, Appendix A, general Design Criteria 1, "Quality Standards and Records"
- C. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

17.3 STANDARDS, CODES, AND GUIDES

It is recommended that the following standards, codes, and guides be considered in establishing a quality assurance program:

- A. Industry Standards and Codes.
 - 1. General QA (applicable to all phases).
 - a. ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities"

b. ANSI N45.2.9-1974, "Requirements for Collection Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants"

c. ANSI N45.2.10-1973, "Quality Assurance Terms and Definitions"

d. ANSI N45.2.12-1977, "Requirements For Auditing of Quality Assurance Programs for Nuclear Power Plants"

e. ANSI N45.2.23-1978, "Qualification of QA Program Audit Personnel for Nuclear Facilities"

f. ANSI Z210.1-1973 (ASTM E380-72), "Metric Practice Guide"

g. ANSI/ASME 626.3-1978, "Qualifications and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code, Section III, Division 1 and 2 Certifying Activities"

h. ASME NQA-1-1979, "Quality Assurance Requirements for Nuclear Power Plants"

2. Design Phase QA.

a. ANSI N45.2.11-1974, "Quality Assurance Requirements for the Design of Nuclear Power Plants"

b. ANSI N45.2.13-1976, "QA Requirements for Control of Procurement of Items and Services of Nuclear Power Plants"

c. ASME Boiler and Pressure Vessel Code, Section III

3. Construction Phase QA.

a. ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants"

b. ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage and Handling Items for Nuclear Power Plants During the Construction Phase"

c. ANSI N45.2.3-1973, "Housekeeping During the Construction Phase of Nuclear Power Plants"

d. ANSI N45.2.3a-1978, "Addenda to ANSI N45.2.3-1973, Housekeeping During the Construction Phase of Nuclear Power Plants"

e. ANSI N45.2.4-1972 (IEEE 336), "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"

f. ANSI/ASME N45.2.5-1978, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete,

Structural Steel, Soils and Foundations During the Construction Phase of Nuclear Plants"

g. ANSI N45.2.6-1978, "Qualification of Inspection, Examination and Testing Personnel for Nuclear Power Plants"

h. ANSI N45.2.8-1975, "Supplementary QA Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants"

i. ANSI N45.2.13-1976, "QA Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

j. ANSI N101.4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities"

k. ANSI N626-1975, "Qualifications and Duties for Authorized Nuclear Inspection"

l. ANSI N626.2-1976, "Qualification and Duties for Authorized Nuclear Inspection (Concrete)"

m. ASME Boiler and Pressure Vessel Code, Sections III and V

4. Operation Phase QA.

a. ANSI N18.7-1976 (ANS 3.2), "Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants"

b. ANSI N626-1974, "Qualification and Duties for Authorized Nuclear Inspection"

c. ANSI-ASME N626.1-1975, "Qualification and Duties for Authorized Nuclear Inservice Inspection"

d. ASME Boiler and Pressure Vessel Code, Section XI

B. RDT Standards.

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

1. General QA (applicable to all phases).

a. F1-2T, "Preparation of System Design Descriptions"

b. F2-2, "Quality Assurance Program Requirements"

2. Design Phase QA.

a. F2-9T, "Reliability Assurance"

3. Construction Phase QA.

- a. E8-12T, "Visual In-Service Inspection System and Associated Equipment for the Reactor Enclosure System"
- b. F2-4T, "Quality Verification Program Requirements"
- c. F2-7T, "Acceptance Sampling Plans"
- d. F3-2T, "Calibration Program Requirements"
- e. F3-7T, "Inspection Requirements for Materials Used in Reactor Coolant Wear Applications"
- f. F3-10T, "Nondestructive Examination of Reactor Core Components"
- g. F3-15T, "Requirements for Inspection of Dimensional Characteristics"
- h. F3-39T, "Testing of High Temperature Cable for Nuclear Detectors"
- i. F5-1T, "Cleaning and Cleanliness Requirements for Nuclear Components"
- j. F7-3T, "Marking of Components and Parts"

4. Operation Phase QA.

- a. F1-4T, "Quality Verification Program Requirements"
- b. F2-7T, "Acceptance Sampling Plans"
- c. F3-2T, "Calibration Program Requirements"

C. Guides.

1. General QA (applicable to all phases).

- a. USNRC Regulatory Guide 1.74, "Quality Assurance Terms and Definitions"
- b. USNRC Regulatory Guide 1.88, "Collection Storage, Maintenance of Nuclear Power Plant Quality Assurance Records"
- c. USNRC Regulatory Guide 1.144, "Auditing of Quality Assurance Programs for Nuclear Power Plants"

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"

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

3. Construction Phase QA.

a. USNRC Regulatory Guide 1.19, "Nondestructive Examination of Primary Containment Liner Welds"

b. USNRC Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)"

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

d. USNRC Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"

e. USNRC Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"

f. USNRC Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants"

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

h. USNRC Regulatory Guide 1.58, "Qualifications of Nuclear Power Plant Inspection, Examination, and Testing Personnel"

i. USNRC Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants"

j. USNRC Regulatory Guide 1.116, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems"

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

4. Operation Phase QA.

a. USNRC Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)"

b. USNRC Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures"

c. USNRC Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"

d. USNRC Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"

e. USNRC Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants"

f. USNRC Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel"

g. USNRC Regulatory Guide 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons"

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

i. USNRC Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations)-Effluent Streams and the Environment"

17.4 SUPPLEMENTARY INFORMATION

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

A. Background Information.

1. USNRC Standard Review Plan 17.1, "Quality Assurance During the Design and Construction Phase,"

2. USNRC Standard Review Plan 17.2, "Quality Assurance During the Operations Phase," *ibid.*

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

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

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

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

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

8. IAEA Safety Standard 50-C-QA, "Quality Assurance in Nuclear Power Plants, A Code of Practice," International Atomic Energy Agency, Vienna

9. IAEA Safety Guide SG-QA1, Draft, "Quality Assurance Program Preparation"

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

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

12. IAEA Safety Guide SG-QA4, Draft, "Quality Assurance for Site Construction of Nuclear Power Plants"

13. IAEA Safety Guide SG-QA5, Draft, "Quality Assurance and Quality Assurance for Operation of Nuclear Power Plants"

14. IAEA Safety Guide SG-QA6, Draft, "Quality Assurance for Design of Nuclear Power Plants"

15. IAEA Safety Guide SG-QA7, Draft, "Quality Assurance Organization for Nuclear Power Plants"

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

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

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

19. IAEA Safety Guide SG-08, Draft, "Surveillance of Important Systems and Components in Nuclear Power Plants"

B. Pending Industry Standards and Codes.

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

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

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

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

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

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

17.5 TOPICS FOR FUTURE STANDARDS DEVELOPMENT

None.

CHAPTER 18

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