Westinghouse Technology Systems Manual

Section 1.1

Reference Documents

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1.1 REFERENCE DOCUMENTS

Learning Objectives:

1. Identify the following reference documents by giving a statement of their contents and/or functions:

   a. Code of Federal Regulations (CFR),
   b. Final Safety Analysis Report (FSAR),
   c. Regulatory Guides (Reg. Guides), and
   d. Technical Specifications (Tech Specs).

2. Define the following terms as stated in the reference documents:

   a. Design Basis,
   b. Reactor Coolant Pressure Boundary,
   c. Loss of Coolant Accident (LOCA),
   d. Single Failure, and
   e. Seismic Category 1.

1.1.1 Introduction

Many data sources were used in the preparation of this manual that provided specific information on the systems and operation of the typical Westinghouse facility. Included in these sources are the Final Safety Analysis Report (FSAR), Westinghouse topical reports (WCAPs), Westinghouse system descriptions, and training manuals from various Westinghouse facilities. Although these documents provide specific system information, there are also documents which provide information related to the minimum requirements for design, operation, and testing of the systems and structures involved at a commercial nuclear facility. Documents included in this group are the Code of Federal Regulations (CFR), Technical Specifications, Regulatory Guides, and various industry standards. The following sections provide a brief description of each of the major documents.

For illustrative purposes, Attachment A to this section contains selected copies of sections of the reference documents described in this chapter.

1.1.2 Code of Federal Regulations

The Code of Federal Regulations (Figure 1.1-2) is a compilation of rules published in the Federal Register by the executive departments and agencies of the Federal Government. The Code of Federal Regulations is kept up to date by the individual issues of the Federal Register. These two publications are used together to determine the latest version of any given rule. Each year a new publication of the code is issued with changes incorporated.

The code is divided into 50 titles which represent broad areas subject to federal regulations. Each title is divided into chapters, which usually bear the names of the issuing agencies. Each chapter is divided into parts covering the specific regulatory areas.

Regulations associated with the Nuclear Regulatory Commission are contained in Title 10 - Energy, Chapter 1 - Nuclear Regulatory Commission, Parts 0 - 199. The regulations are cited using the title, part, section, and paragraph designations (Figure 1.1-3). For example, 10 CFR 50.34(b) refers to Title 10 of the Code of Federal Regulations, Part 50, Section 34, paragraph (b).

The following is a list and brief description of the parts of 10CFR that primarily apply to NRC licensed commercial nuclear reactors (Figure 1.1-4):

- **Part 2** Policy and procedures related to issuing, amending, or revoking an operating license; enforcement actions; and public rule making.
- **Part 19** Requirements for disseminating information to nuclear plant workers concerning radiological working conditions, enforcement actions, etc. Rules of conduct for NRC inspections.**Part 20** Standards for protection against radiation
- **Part 21** Reporting of defects and noncompliance
- **Part 50** Rules for license application, content of applications, facility design requirements, and reporting of events to the NRC.

  Appendix A - General Design Criteria
  Appendix B - Quality Assurance Criteria

- **Part 55** Rules and procedures for the licensing of reactor operators.
- **Part 71** Requirements for packaging, shipping and transportation of radioactive material.
- **Part 73** Requirements related to physical protection of the facility to protect against radiological sabotage and theft of special nuclear material.
- **Part 100** Reactor site criteria including population density and seismic and geologic evaluations.

  Appendix A - Seismic and Geologic Siting Criteria for Nuclear Power Plants

Included in all these parts are definitions of terms important to understanding the regulations. For example, the following terms are defined in 10 CFR 50 (Figure 1.1-5):

1. “Design basis” means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design.

2. “Reactor coolant system pressure boundary” means all those pressure-containing components of water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves which are: (1) part of the reactor coolant system, or (2) connected to the reactor coolant system, up to and including (a) the outermost containment isolation valve in system piping which penetrates primary reactor containment, (b) the second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment, (c) the reactor coolant system safety and relief valves.“Loss of coolant accident” means a postulated accident that results from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

3. "Single failure" means an occurrence which results in the loss of capability of a component to perform its intended safety functions. A system is considered to be designed against an assumed single failure if neither (1) a single failure of any active component nor (2) a single failure of any passive component results in a loss of the capability of the system to perform its safety functions.

4. "Safe shutdown earthquake" is defined in 10 CFR 100 as that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology. The "safe shutdown earthquake" defines that earthquake which has commonly been referred to as the "design basis earthquake." It is that earthquake for which certain structures, systems, and components are designed to remain functional.

1.1.3 Final Safety Analysis Report (FSAR)

A Final Safety Analysis Report (FSAR) is submitted with each application for an operating license and includes a description of the facility, the design bases and limits on its operation, and a safety analysis of the structures, systems, and components of the facility. The function of the FSAR is to demonstrate the applicant's qualifications, capability, and planned controls to assure safe plant operation within the constraints of plant design, operating limitations and regulatory requirements. (See Figure 1.1-7.)

The requirement for having an FSAR and the minimum information required to be included in it is established in 10 CFR 50.34(b). (See Attachment A.) For example, this regulation, in part, requires an evaluation and analysis of the emergency core cooling system (ECCS) cooling performance following postulated loss-of-coolant accidents to ensure that the requirements of 10 CFR 50.46, "ECCS Design Acceptance Criteria," are met. This analysis is included in FSAR Chapter 15, "Accident Analysis," along with evaluations to show safe plant response for other postulated normal and abnormal plant conditions. Other examples of information contained in the FSAR include the methods with which the licensee plans to meet the 10 CFR 50 Appendix B Quality Assurance Criteria and the results of

environmental and meteorology monitoring programs as they pertain to 10 CFR 100 requirements. The plant is required to maintain the FSAR current and to submit the most up-to-date version to the NRC on a yearly basis (commonly referred to as the Updated FSAR or UFSAR).

1.1.4 Technical Specifications

The requirement for including Technical Specifications (Figure 1.1-8) as part of the license application is set forth in 10 CFR 50.36. (See Attachment A.) The NRC-approved Technical Specifications are issued to the facility as part of the operating license. (See Attachment A for an example of a facility operating license.) The Technical Specifications establish minimum operating limits for the facility. Failure to comply with these limits may require the reduction of the allowable operating power level or, in some cases, even a complete shutdown and cooldown of the unit.

The bases for the operating limits established in Technical Specifications are the analyses and evaluations included in the FSAR. Operating within the established limits ensures that the assumptions made in the safety analyses are true for all operating conditions. Technical Specifications are required to include the following sections:

1. Safety limits and limiting safety system settings. Safety limits are limits upon important process variables which are found to be necessary to protect the integrity of the physical barriers which guard against the uncontrolled release of radioactivity to the environment. If any safety limit is exceeded, the reactor shall be shutdown.

   Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting will assure that automatic protective action will correct the abnormal situation before a safety limit is exceeded. Appropriate action for exceeding a limiting safety system setting may include shutting down the reactor.

2. Limiting conditions for operation (LCOs) are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO is exceeded, remedial action is required within a specified time frame. (See Attachment A for an example of a Technical Specification LCO.)

3. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary system or component quality is maintained.
4. Design features are those features of the facility such as materials of construction and geometric arrangements which, if altered or modified, would have a significant effect on safety and are not covered in Sections 1-3.

5. Administrative controls are provisions related to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner.

A bases section is included with the Technical Specifications as part of a facility's operating license application, as required by 10 CFR 50.36, and provides the reason(s) for each individual specification. (See Attachment A for an example of a Technical Specification basis entry.) The bases section is included with the Technical Specifications for information, but is not part of the Technical Specifications.

1.1.5 Codes and Standards

Since the CFR is written in general terms, supplementary documentation is necessary to further define the requirements stated in the CFR. (See Figure 1.1-9.) Each FSAR contains a list of the specific codes and standards to which a particular licensee has committed to fulfill regulatory obligations. The following sections describe three of the most used documents and include examples of each.

1.1.5.1 American National Standards Institute (ANSI) Standards

ANSI Standards cover a wide range of subjects. Certain ANSI standards were written to amplify the general design criteria of 10 CFR 50, Appendix A. For example, ANSI Standard 18.2 defines the following design condition categories:

- Condition I - Normal Operation
- Condition II - Incidents of Moderate Frequency
- Condition III - Infrequent Incidents
- Condition IV - Limiting Faults

ANSI 18.2 defines each condition by the expected frequency of occurrence and its probability of deteriorating to a worse condition. Design requirements for each condition are based on the amount of resulting core damage and radioactive release permitted. These design conditions and requirements are analyzed for each plant, and the results are documented in the facility's FSAR. (See Attachment A.)

ANSI Standard 18.2a defines safety classes used to designate safety systems and components in accordance with their importance to nuclear safety. ANSI 18.2a defines a safety system as any system that is necessary to shut down the reactor, cool the core, cool another safety system, or cool the reactor containment after an accident. In addition, any system that contains, controls, or reduces radioactivity released in an accident is a safety system. Safety Class 1 applies to components whose failure could cause a Condition III or Condition IV loss-of-reactor-coolant accident. Safety Class 2 generally applies to reactor containment and RCS pressure boundary components not in Safety Class 1. Also included in Safety Class 2 are safety systems that remove heat from the reactor or reactor containment, circulate reactor coolant, or control radioactivity or hydrogen in containment. The last two safety classes, Safety Class 3 and Non-nuclear Safety Class, apply to other

plant components related to safe plant operation or the potential uncontrolled release of radioactivity.

1.1.5.2 American Society of Mechanical Engineers (ASME) Code

The ASME boiler and pressure vessel code is used to provide design criteria for fabrication, inspection, and construction of systems and vessels. The two most referenced sections with regard to nuclear plant systems are sections III and XI. These sections are discussed below.

Section III, Rules for Construction of Nuclear Power Plant Components. The rules of this section constitute requirements for the design, construction, stamping, and overpressure protection of nuclear power plant items such as vessels, concrete reactor vessels and concrete containments, storage tanks, piping systems, pumps, valves, core support structures, and component supports for use in, or containment of, portions of the nuclear power system of any power plant.

Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components. The rules of this section constitute requirements for inservice inspection, nondestructive examination (NDE), and testing of pumps and valves in nuclear power plants. This section defines such items as the required NDE of components or welds; allowable valve stroke times; and tolerances on pump flow, discharge pressure, and vibration. Requirements for the use of this section are contained in the plant technical specifications.

The ASME code also classifies components according to their use and importance to nuclear safety. Code Classes 1 - 3 correspond to ANSI 18.2a safety classes, with the exception of reactor containment components, which are designated Code Class MC. These classifications specify design and quality assurance requirements.

The ASME code is frequently revised. To know which editions and addenda are required for a particular facility, 10 CFR 50.55a defines applicability according to the issue date of the facility's construction permit. In addition, the FSAR contains information on the codes and standards that are followed during the design and construction of plant systems.

1.1.5.3 Institute of Electrical and Electronic Engineers (IEEE) Standards

IEEE standards are used in the design, operation, and testing of nuclear power plant electrical and instrumentation components and systems. Some of the standards developed by the IEEE are listed below.

1. Criteria for Protection Systems for Nuclear Power Generating Stations,
2. Guide to the Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems,
3. Guide for Qualification Testing of Nuclear Power Plant Protection Systems,
4. Guide for Qualification of Engineered Safety Feature Motors for Nuclear Fueled Generating Stations,

5. Guide for Qualification Testing of Electrical Cables Used in Nuclear Power Plants, and
6. Guide for Qualification Testing of Electrical Penetrations in Nuclear Plant Containments.

IEEE standards define as Class 1E, electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or are otherwise essential in preventing significant release of radioactive material to the environment.

1.1.6 Regulatory Guides

NRC Regulatory Guides (Figure 1.1-10) were formerly called Safety Guides. They are not legal documents or requirements. However, they make available to the public methods acceptable to the NRC staff for complying with specific portions of 10 CFR. In some cases a Regulatory Guide endorses an industry standard wholly or in part.

Applications for the use of Regulatory Guides are as follows:

1. To amplify the Code of Federal Regulations,
2. To endorse and/or supplement industry standards, or
3. To provide guidance in ensuring specific regulatory requirements are met.

Each Regulatory Guide consists of four parts:

1. **Introduction** - References to applicable codes, standards, and Code of Federal Regulations associated with a particular subject.

2. **Discussion** - Information on the development of standards associated with the subject. The discussion may address areas of disagreement, if any exists, concerning those standards.

3. **Regulatory Position** - Definitions acceptable to the NRC, and criteria, the basis for the criteria, and any additional information required to establish the NRC's position on the particular subject.

4. **Implementation** - A discussion of the NRC staff use of the Regulatory Guide and any alternative methods acceptable for fulfilling the requirements discussed in the Regulatory Guide.

For example, Regulatory Guide 1.29, "Seismic Design Classification," (see Attachment A) amplifies 10 CFR 50 and 10 CFR 100 requirements. This guide also provides the definition of the term “Seismic Category I.” Seismic Category I refers to those plant structures, systems and components which are important to safety and are designed to remain functional in the event of a "Safe Shutdown Earthquake." Seismic Category I structures, systems, and components are necessary to assure the:

1. Integrity of the reactor coolant pressure boundary,
2. Capability to shut down the reactor and maintain it in a safe shutdown condition, or
3. Capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10CFR100.

1.1.7 Summary

The interrelationships between the various reference documents can be illustrated using seismic design considerations as an example (Figure 1.1-11). General Design Criterion 2 of 10 CFR 50, Appendix A (see Attachment A) requires that certain systems be designed for protection against natural phenomena such as earthquakes. 10 CFR 100, Appendix A (also in Attachment A) provides more specific requirements regarding the evaluations and analyses that must be done to ensure adequate seismic suitability of the site and design of the plant. These required evaluations and analyses are documented in the plant's FSAR. Individual Technical Specifications set forth the associated operating requirements to ensure that plant parameters are monitored and plant systems function as assumed in the FSAR, with the bases section tying the particular specification back to the FSAR analyses. (See Attachment A.) Any Regulatory Guides or industry standards that were used in the evaluation process may be referenced in the FSAR discussion and/or the Technical Specification bases.

REFERENCE DOCUMENTS

CODE OF FEDERAL REGULATIONS

FINAL SAFETY ANALYSIS REPORT

TECHNICAL SPECIFICATIONS

OPERATING LICENSE

REGULATORY GUIDES

AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) BOILER AND PRESSURE VESSEL CODE

INSTITUTE OF ELECTRICAL AND ELECTRONICS ENGINEERS (IEEE) STANDARDS

AMERICAN NATIONAL STANDARDS INSTITUTE (ANSI) STANDARDS

Figure 1.1-1 Reference Documents

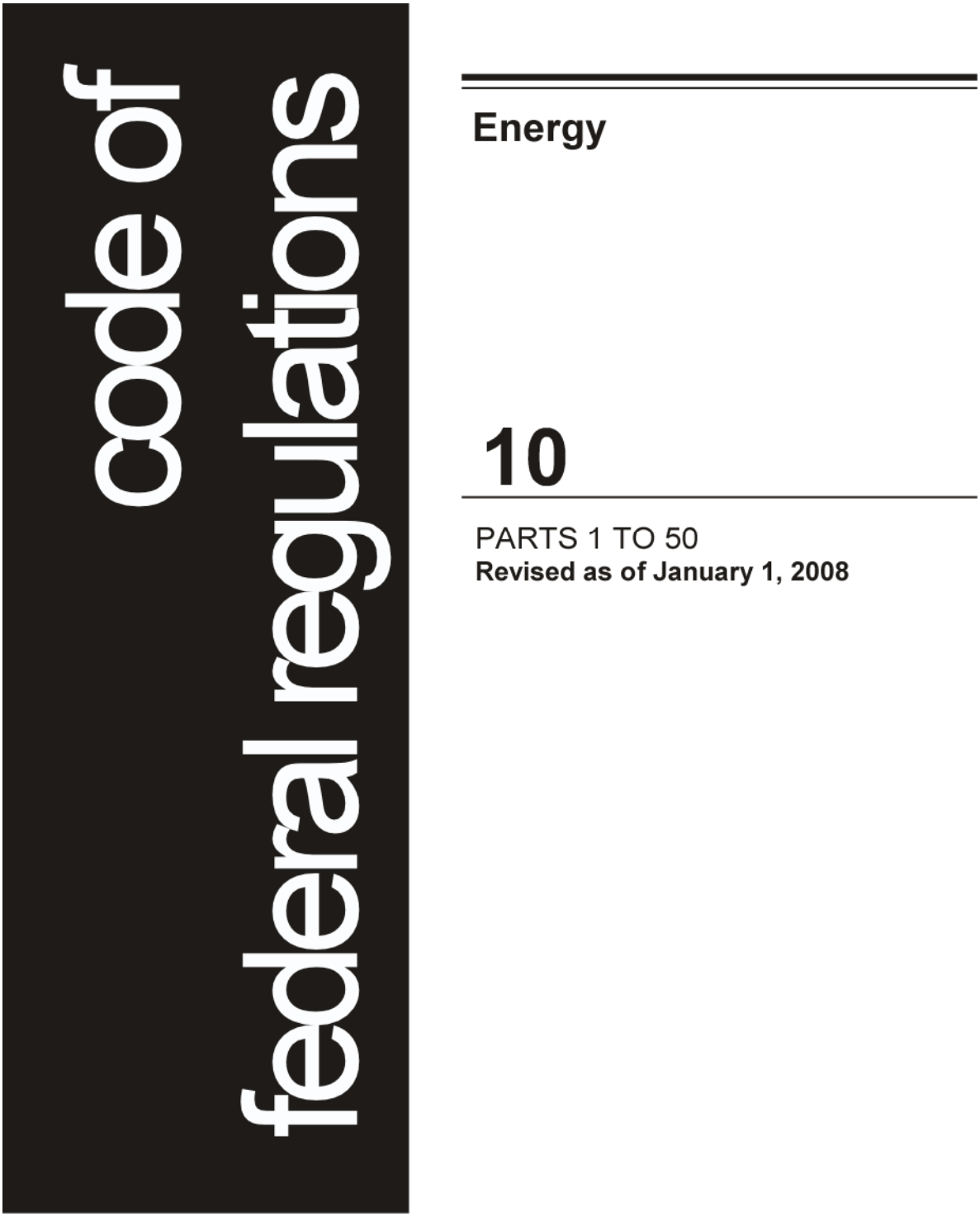

Figure 1.1-2 Code of Federal Regulations

10 CFR 50.34(b)

Figure 1.1-3 Title, Chapter, Part

TITLE 10
CODE OF FEDERAL REGULATIONS

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Figure 1.1-4 Title 10, Table of Contents

10 CFR 50

DEFINITIONS

- DESIGN BASIS
- REACTOR COOLANT PRESSURE BOUNDARY
- LOSS OF COOLANT ACCIDENT (LOCA)
- SINGLE FAILURE

Figure 1.1-5 10 CFR 50 Definitions

10 CFR 50

LICENSE REQUIREMENTS

50.10: LICENSE REQUIRED

50.20: CLASSES OF LICENSES
POWER REACTORS
MATERIALS (Medical, R&D)

50.34: CONTENTS OF APPLICATIONS; TECHNICAL INFORMATION

(a) Preliminary Safety Analysis Report (PSAR)
- submitted with application for construction permit

(b) Final Safety Analysis Report (FSAR)
- submitted with application for operating license

Figure 1.1-6 10 CFR 50 Requirements

FSAR

Description and Safety Assessment of Site (10 CFR 100)

Description of the Facility Design and Design Bases
(10 CFR 50, Appendix A - ANSI 18.2A, Safety Classes)

Accident Analysis (10 CFR 50.46, ECCS Acceptance Criteria)
- Condition I - Normal Operation and Operational Transients
- Condition II - Faults of Moderate Frequency
- Condition III - Infrequent Faults
- Condition IV - Limiting Faults
  - (ANSI 18.2, Conditions for Design)

Technical Specification (10 CFR 50.36)

Description of Quality Assurance Program (10 CFR 50, Appendix B)

Other information (10 CFR 50.34(b))

Figure 1.1-7 Final Safety Analysis Report

10 CFR 50.36
TECHNICAL SPECIFICATIONS

Chapter 16 of FSAR

Validate assumptions made in FSAR analysis
Technical Specifications are Appendix A to the plants Operating License

The following must be included the Technical Specifications (10 CFR 50.36(c))

SAFETY LIMITS
LIMITING SAFETY SYSTEM SETTINGS
LIMITING CONDITIONS FOR OPERATION
SURVEILLANCE REQUIREMENTS
DESIGN FEATURES
ADMINISTRATIVE CONTROLS

Figure 1.1-8 Technical Specifications

10 CFR 50.55a
CODES AND STANDARDS

Incorporates by reference sections of the ASME Boiler & Pressure Vessel Code for Design, Construction, and Testing related to:

PRESSURE VESSELS
PIPING
PUMPS
VALVES
IN-SERVICE INSPECTION

Incorporates by reference IEEE Standard 279-1971 for Design and Testing of:

PROTECTION SYSTEMS

Figure 1.1-9 Codes and Standards

U. S. NUCLEAR REGULATORY COMMISSION REGULATORY GUIDES

Amplifies the Code of Federal Regulations

Endorses/supplements Industry Standards

Provides guidance and/or additional information to the code

REGULATORY GUIDE contains the following:

A. Introduction
B. Discussion
C. Regulatory Position
D. Implementation

Figure 1.1-10 Regulatory Guides

10 CFR 100
REACTOR SITE CRITERIA

10 CFR 100, Appendix A, Seismic and Geologic Siting Criteria

SAFE SHUTDOWN EARTHQUAKE
OPERATING BASIS EARTHQUAKE

10 CFR 50, Appendix A, General Design Criterion 2
Design basis for protection against natural phenomena

Regulatory Guide 1.29 - Seismic Design Classification
Seismic Category I

Figure 1.1-10 Regulatory Guides

Section 1.1, Attachment A

Passages from Selected Regulatory Documents

10 CFR 34 - Contents of construction permit and operating license applications; technical information.

(a) *Preliminary safety analysis report*. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information[5] to be included shall consist of the following:

(1) Stationary power reactor applicants for a construction permit who apply on or after January 10, 1997, shall comply with paragraph (a)(1)(ii) of this section. All other applicants for a construction permit shall comply with paragraph (a)(1)(i) of this section.

(i) A description and safety assessment of the site on which the facility is to be located, with appropriate attention to features affecting facility design. Special attention should be directed to the site evaluation factors identified in part 100 of this chapter. The assessment must contain an analysis and evaluation of the major structures, systems and components of the facility which bear significantly on the acceptability of the site under the site evaluation factors identified in part 100 of this chapter, assuming that the facility will be operated at the ultimate power level which is contemplated by the applicant. With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by this paragraph, in support of the application for a construction permit, or a design approval.

(ii) A description and safety assessment of the site and a safety assessment of the facility. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The following power reactor design characteristics and proposed operation will be taken into consideration by the Commission:

(A) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(B) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(C) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(D) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In

performing this assessment, an applicant shall assume a fission product release[6] from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with applicable site characteristics, including site meteorology, to evaluate the offsite radiological consequences. Site characteristics must comply with part 100 of this chapter. The evaluation must determine that:

(1) An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 25 rem[7] total effective dose equivalent (TEDE).

(2) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE);

(E) With respect to operation at the projected initial power level, the applicant is required to submit information prescribed in paragraphs (a)(2) through (a)(8) of this section, as well as the information required by paragraph (a)(1)(i) of this section, in support of the application for a construction permit.

(2) A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility.[8] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement, and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

(4) A preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing

the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. Analysis and evaluation of ECCS cooling performance and the need for high point vents following postulated loss-of-coolant accidents must be performed in accordance with the requirements of § 50.46 and § 50.46a of this part for facilities for which construction permits may be issued after December 28, 1974.

(5) An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of technical specifications for the facility, with special attention given to those items which may significantly influence the final design: *Provided, however*, That this requirement is not applicable to an application for a construction permit filed prior to January 16, 1969.

(6) A preliminary plan for the applicant's organization, training of personnel, and conduct of operations.

(7) A description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for quality assurance programs for nuclear power plants and fuel reprocessing plants. The description of the quality assurance program for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of appendix B will be satisfied.

(8) An identification of those structures, systems, or components of the facility, if any, which require research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility.

(9) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(10) A discussion of the applicant's preliminary plans for coping with emergencies. Appendix E sets forth items which shall be included in these plans.

(11) On or after February 5, 1979, applicants who apply for construction permits for nuclear powerplants to be built on multiunit sites shall identify potential hazards to

the structures, systems and components important to safety of operating nuclear facilities from construction activities. A discussion shall also be included of any managerial and administrative controls that will be used during construction to assure the safety of the operating unit.

(12) On or after January 10, 1997, stationary power reactor applicants who apply for a construction permit, as partial conformance to General Design Criterion 2 of appendix A to this part, shall comply with the earthquake engineering criteria in appendix S to this part.

(b) *Final safety analysis report*. Each application for an operating license shall include a final safety analysis report. The final safety analysis report shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and shall include the following:

(1) All current information, such as the results of environmental and meteorological monitoring programs, which has been developed since issuance of the construction permit, relating to site evaluation factors identified in part 100 of this chapter.

(2) A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefore, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

(i) For nuclear reactors, such items as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.

(ii) For facilities other than nuclear reactors, such items as the chemical, physical, metallurgical, or nuclear process to be performed, instrumentation and control systems, ventilation and filter systems, electrical systems, auxiliary and emergency systems, and radioactive waste handling systems shall be discussed insofar as they are pertinent.

(3) The kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20 of this chapter.

(4) A final analysis and evaluation of the design and performance of structures, systems, and components with the objective stated in paragraph (a)(4) of this section and taking into account any pertinent information developed since the

submittal of the preliminary safety analysis report. Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of § 50.46 for facilities for which a license to operate may be issued after December 28, 1974.

(5) A description and evaluation of the results of the applicant's programs, including research and development, if any, to demonstrate that any safety questions identified at the construction permit stage have been resolved.

(6) The following information concerning facility operation:

(i) The applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements.

(ii) Managerial and administrative controls to be used to assure safe operation. Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the requirements for such controls for nuclear power plants and fuel reprocessing plants. The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of appendix B will be satisfied.

(iii) Plans for preoperational testing and initial operations.

(iv) Plans for conduct of normal operations, including maintenance, surveillance, and periodic testing of structures, systems, and components.

(v) Plans for coping with emergencies, which shall include the items specified in appendix E.

(vi) Proposed technical specifications prepared in accordance with the requirements of § 50.36.

(vii) On or after February 5, 1979, applicants who apply for operating licenses for nuclear powerplants to be operated on multiunit sites shall include an evaluation of the potential hazards to the structures, systems, and components important to safety of operating units resulting from construction activities, as well as a description of the managerial and administrative controls to be used to provide assurance that the limiting conditions for operation are not exceeded as a result of construction activities at the multiunit sites.

(7) The technical qualifications of the applicant to engage in the proposed activities in accordance with the regulations in this chapter.

(8) A description and plans for implementation of an operator requalification program. The operator requalification program must as a minimum, meet the requirements for those programs contained in § 55.59 of part 55 of this chapter.

(9) A description of protection provided against pressurized thermal shock events, including projected values of the reference temperature for reactor vessel beltline materials as defined in § 50.61 (b)(1) and (b)(2).

(10) On or after January 10, 1997, stationary power reactor applicants who apply for an operating license, as partial conformance to General Design Criterion 2 of appendix A to this part, shall comply with the earthquake engineering criteria of appendix S to this part. However, for those operating license applicants and holders whose construction permit was issued before January 10, 1997, the earthquake engineering criteria in Section VI of appendix A to part 100 of this chapter continues to apply.

(11) On or after January 10, 1997, stationary power reactor applicants who apply for an operating license, shall provide a description and safety assessment of the site and of the facility as in § 50.34(a)(1)(ii). However, for either an operating license applicant or holder whose construction permit was issued before January 10, 1997, the reactor site criteria in part 100 of this chapter and the seismic and geologic siting criteria in appendix A to part 100 of this chapter continues to apply.

(c) *Physical Security Plan*. Each application for an operating license for a production or utilization facility must include a physical security plan. The plan must describe how the applicant will meet the requirements of part 73 of this chapter (and part 11 of this chapter, if applicable, including the identification and description of jobs as required by § 11.11(a) of this chapter, at the proposed facility). The plan must list tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR parts 11 and 73, if applicable.

(d) *Safeguards contingency plan*. Each application for an operating license for a production or utilization facility that will be subject to §§ 73.50, 73.55, or § 73.60 of this chapter, must include a licensee safeguards contingency plan in accordance with the criteria set forth in appendix C to 10 CFR part 73. The safeguards contingency plan shall include plans for dealing with threats, thefts, and radiological sabotage, as defined in part 73 of this chapter, relating to the special nuclear material and nuclear facilities licensed under this chapter and in the applicant's possession and control. Each application for such a license shall include the first four categories of information contained in the applicant's safeguards contingency plan. (The first four categories of information as set forth in appendix C to 10 CFR part 73 of this chapter are Background, Generic Planning Base, Licensee Planning Base, and Responsibility Matrix. The fifth category of information, Procedures, does not have to be submitted for approval.)[9]

(e) *Protection against unauthorized disclosure*. Each applicant for an operating license for a production or utilization facility, who prepares a physical security plan, a safeguards contingency plan, or a guard qualification and training plan, shall protect the plans and other related safeguards information against unauthorized disclosure in accordance with the requirements of § 73.21 of this chapter, as appropriate.

(f) *Additional TMI-related requirements*. In addition to the requirements of paragraph (a) of this section, each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982, shall meet the requirements in paragraphs (f)(1) through (3) of this section. This regulation applies to the pending applications by Duke Power Company (Perkins Nuclear Station, Units 1, 2, and 3), Houston Lighting & Power Company (Allens Creek Nuclear Generating Station, Unit 1), Portland General Electric Company (Pebble Springs Nuclear Plant, Units 1 and 2), Public Service Company of Oklahoma (Black Fox Station, Units 1 and 2), Puget Sound Power & Light Company (Skagit/Hanford Nuclear Power Project, Units 1 and 2), and Offshore Power Systems (License to Manufacture Floating Nuclear Plants). The number of units that will be specified in the manufacturing license above, if issued, will be that number whose start of manufacture, as defined in the license application, can practically begin within a 10-year period commencing on the date of issuance of the manufacturing license, but in no event will that number be in excess of ten. The manufacturing license will require the plant design to be updated no later than 5 years after its approval. Paragraphs (f)(1)(xii), (2)(ix), and (3)(v) of this section, pertaining to hydrogen control measures, must be met by all applicants covered by this regulation. However, the Commission may decide to impose additional requirements and the issue of whether compliance with these provisions, together with 10 CFR 50.44 and criterion 50 of appendix A to 10 CFR part 50, is sufficient for issuance of that manufacturing license which may be considered in the manufacturing license proceeding. In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section, except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

(1) To satisfy the following requirements, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of these studies are factored into the final design of the facility. For licensees identified in the introduction to paragraph (f) of this section, all studies must be completed no later than 2 years following the issuance of the construction permit or manufacturing license.[10] For all other applicants, the studies must be submitted as part of the final safety analysis report.

(i) Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (II.B.8)

(ii) Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's only) (II.E.1.1):

(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(B) A design review of AFWS.

(C) An evaluation of AFWS flow design bases and criteria.

(iii) Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)

(iv) Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (II.K.3.2)

(v) Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.13)

(vi) Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)

(vii) Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)

(viii) Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)

(ix) Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-

pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI") (Applicable to BWR's only). (II.K.3.24)

(x) Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)

(xi) Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only) (II.K.3.45)

(xii) Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include:

(A) A comparison of costs and benefits of the alternative systems considered.

(B) For the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section.

(C) For the selected system, preliminary design descriptions of equipment, function, and layout.

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues.

(i) Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only) (I.A.4.2.)

(ii) Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)

(iii) Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)

(iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)

(v) Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)

(vi) Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)

(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term1A[11] radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term1A[11] radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)

(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that: (II.B.8)

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10% during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100% fuel clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation.

(x) Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)

(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)

(xii) Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)

(xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)

(xiv) Provide containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(C) Do not result in reopening of the containment isolation valves on resetting of the isolation signal,

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)

(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)

(xvii) Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)

(xviii) Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)

(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)

(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)

(xxi) Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)

(xxii) Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only). (II.K.2.9)

(xxiii) Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only). (II.K.2.10)

(xxiv) Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23)

(xxv) Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (III.A.1.2).

(xxvi) Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term1A[11] radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)

(xxvii) Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)

(xxviii) Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term[11] release, and make necessary design provisions to preclude such problems. (III.D.3.4)

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy paragraph (a)(1) of this section or to address the applicant's technical qualifications and management structure and competence.

(i) Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)

(ii) Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR part 50 includes all structures, systems, and components important to safety. (I.F.1)

(iii) Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality

assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a QA role in design and analysis activities. (I.F.2)

(iv) Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)

(v) Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8)

(A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 Subsubarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraphs (f)(3)(v)(A)(1) and (f)(3)(v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. It is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852-2738.

(B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not

including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsubarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsubarticle CC-3720, Service Load Category, (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.

(vi) For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)

(vii) Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)

(g) *Combustible gas control.* All applicants for a reactor construction permit or operating license whose application is submitted after October 16, 2003, shall include the analyses, and the descriptions of the equipment and systems required by § 50.44 as a part of their application.

(h) *Conformance with the Standard Review Plan (SRP).* (1)(i) Applications for light water cooled nuclear power plant operating licenses docketed after May 17, 1982 shall include an evaluation of the facility against the Standard Review Plan (SRP) in effect on May 17, 1982 or the SRP revision in effect six months prior to the docket date of the application, whichever is later.

(ii) Applications for light-watercooled nuclear power plant construction permits docketed after May 17, 1982, shall include an evaluation of the facility against the SRP in effect on May 17, 1982, or the SRP revision in effect six months before the docket date of the application, whichever is later.

(2) The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for a facility and those corresponding features,

techniques, and measures given in the SRP acceptance criteria. Where such a difference exists, the evaluation shall discuss how the alternative proposed provides an acceptable method of complying with those rules or regulations of Commission, or portions thereof, that underlie the corresponding SRP acceptance criteria.

(3) The SRP was issued to establish criteria that the NRC staff intends to use in evaluating whether an applicant/licensee meets the Commission's regulations. The SRP is not a substitute for the regulations, and compliance is not a requirement. Applicants shall identify differences from the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP criteria provide an acceptable method of complying with the Commission's regulations.

[33 FR 18612, Dec. 17, 1968; 72 FR 49491, Aug. 28, 2007]

[5] The applicant may provide information required by this paragraph in the form of a discussion, with specific references, of similarities to and differences from, facilities of similar design for which applications have previously been filed with the Commission.

[6] The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.

[7] A whole body dose of 25 rem has been stated to correspond numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations at the time could be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, its use is not intended to imply that this number constitutes an acceptable limit for an emergency dose to the public under accident conditions. Rather, this dose value has been set forth in this section as a reference value, which can be used in the evaluation of plant design features with respect to postulated reactor accidents, in order to assure that such designs provide assurance of low risk of public exposure to radiation, in the event of such accidents.

[8] General design criteria for chemical processing facilities are being developed.

[9] A physical security plan that contains all the information required in both § 73.55 and appendix C to part 73 of this chapter satisfies the requirement for a contingency plan.

[10]Alphanumeric designations correspond to the related action plan items in NUREG 0718 and NUREG–0660, "NRC Action Plan Developed as a Result of the TMI–2 Accident." They are provided herein for information only.

[11] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

10 CFR 50.36 - Technical specifications.

(a) Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

(b) Each license authorizing operation of a production or utilization facility of a type described in § 50.21 or § 50.22 will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34. The Commission may include such additional technical specifications as the Commission finds appropriate.

(c) Each applicant for a design certification or manufacturing license under part 52 of this chapter shall include in its application proposed generic technical specifications in accordance with the requirements of this section for the portion of the plant that is within the scope of the design certification or manufacturing license application.

(d) Technical specifications will include items in the following categories:

(1) *Safety limits, limiting safety system settings, and limiting control settings*. (i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor, except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.

(B) Safety limits for fuel reprocessing plants are those bounds within which the process variables must be maintained for adequate control of the operation and that must not be exceeded in order to protect the integrity of the physical system that is designed to guard against the uncontrolled release or radioactivity. If any safety limit for a fuel reprocessing plant is exceeded, corrective action must be taken as stated in the technical specification or the affected part of the process, or the entire process if required, must be shut down, unless this action would further reduce the margin of

safety. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. If a portion of the process or the entire process has been shutdown, operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.

(ii)(A) Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.

(B) Limiting control settings for fuel reprocessing plants are settings for automatic alarm or protective devices related to those variables having significant safety functions. Where a limiting control setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that protective action, either automatic or manual, will correct the abnormal situation before a safety limit is exceeded. If, during operation, the automatic alarm or protective devices do not function as required, the licensee shall take appropriate action to maintain the variables within the limiting control-setting values and to repair promptly the automatic devices or to shut down the affected part of the process and, if required, to shut down the entire process for repair of automatic devices. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the plant.

(2) *Limiting conditions for operation*. (i) Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation

or follow any remedial action permitted by the technical specifications until the condition can be met. In the case of a nuclear reactor not licensed under § 50.21(b) or § 50.22 of this part or fuel reprocessing plant, the licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. The licensee shall retain the record of the results of each review until the Commission terminates the license for the nuclear reactor or the fuel reprocessing plant. In the case of nuclear power reactors licensed under § 50.21(b) or § 50.22, the licensee shall notify the Commission if required by § 50.72 and shall submit a Licensee Event Report to the Commission as required by § 50.73. In this case, licensees shall retain records associated with preparation of a Licensee Event Report for a period of three years following issuance of the report. For events which do not require a Licensee Event Report, the licensee shall retain each record as required by the technical specifications.

(ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

(A) *Criterion 1*. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) *Criterion 2*. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(C) *Criterion 3*. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

(D) *Criterion 4*. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

(iii) A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of this section.

(3) *Surveillance requirements*. Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

(4) *Design features*. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or

modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c) (1), (2), and (3) of this section.

(5) *Administrative controls*. Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4.

(6) *Decommissioning*. This paragraph applies only to nuclear power reactor facilities that have submitted the certifications required by § 50.82(a)(1) and to non-power reactor facilities which are not authorized to operate. Technical specifications involving safety limits, limiting safety system settings, and limiting control system settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls will be developed on a case-by-case basis.

(7) *Initial notification*. Reports made to the Commission by licensees in response to the requirements of this section must be made as follows:

(i) Licensees that have an installed Emergency Notification System shall make the initial notification to the NRC Operations Center in accordance with §50.72 of this part.

(ii) All other licensees shall make the initial notification by telephone to the Administrator of the appropriate NRC Regional Office listed in appendix D, part 20, of this chapter.

(8) *Written Reports*. Licensees for nuclear power reactors licensed under § 50.21(b) and § 50.22 of this part shall submit written reports to the Commission in accordance with § 50.73 of this part for events described in paragraphs (c)(1) and (c)(2) of this section. For all licensees, the Commission may require Special Reports as appropriate.

(e)(1) This section shall not be deemed to modify the technical specifications included in any license issued prior to January 16, 1969. A license in which technical specifications have not been designated shall be deemed to include the entire safety analysis report as technical specifications.

(2) An applicant for a license authorizing operation of a production or utilization facility to whom a construction permit has been issued prior to January 16, 1969, may submit technical specifications in accordance with this section, or in accordance with the requirements of this part in effect prior to January 16, 1969.

(3) At the initiative of the Commission or the licensee, any license may be amended to include technical specifications of the scope and content which would be required if a new license were being issued.

(f) The provisions of this section apply to each nuclear reactor licensee whose authority to operate the reactor has been removed by license amendment, order, or regulation.

Facility Operating License

LICENSE AUTHORITY FILE COPY

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

Full Power License issued 10-18-84

UNION ELECTRIC COMPANY

DOCKET NO. STN 50-483

CALLAWAY PLANT UNIT NO. 1

FACILITY OPERATING LICENSE

License No. NPF-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for license filed by Union Electric Company* (licensee), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Callaway Plant, Unit No. 1 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-139 and the application, as amended, the provisions of the Act, and the regulations of the Commission;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   E. Union Electric Company is technically qualified to engage in the activities authorized by this license in accordance with the Commission's regulations set forth in 10 CFR Chapter I;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140 "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

*As of the closing of the Merger contemplated by the Agreement and Plan of Merger, by and among Union Electric Company, CIPSCO Incorporated, Ameren Corporation and Arch Merger, Inc., dated August 11, 1995, Union Electric Company is a wholly-owned operating subsidiary of Ameren Corporation.

Amendment No. 120
FEB 13 1998

H. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of this Facility Operating License No. NPF-30, subject to the conditions for protection of the environment set forth in the Environmental Protection Plan attached as Appendix B, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70.

2. Pursuant to approval by the Nuclear Regulatory Commission at a meeting on October 4, 1984, the License for Fuel Loading and Low Power Testing, License No. NPF-25, issued on June 11, 1984, is superseded by Facility Operating License No. NPF-30 hereby issued to Union Electric (UE) to read as follows:

A. The license applies to the Callaway Plant, Unit No. 1, a pressurized water nuclear reactor and associated equipment (the facility), owned by Union Electric Company. The facility is located in central Missouri within Callaway County, Missouri, and is described in the licensee's "Final Safety Analysis Report", as supplemented and amended, and in the licensee's Environmental Report, as supplemented and amended.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Union Electric Company (UE):

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities," UE to possess, use and operate the facility at the designated location in Callaway County, Missouri, in accordance with the procedures and limitations set forth in this license;

(2) UE, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) UE, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

UE is authorized to operate the facility at reactor core power levels not in excess of 3565 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan*

The Technical Specifications contained in Appendix A, as revised through Amendment No. 184 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Environmental Qualification (Section 3.11, SSER #3) **

Deleted per Amendment No. 169

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* Amendments 133, 134, &135 were effective as of April 30, 2000 however these amendments were implemented on April 1, 2000.

** The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

(4) Surveillance of Hafnium Control Rods (Section 4.2.3.1(10), SER and SSER #2)

Deleted per Amendment No. 169

(5) Fire Protection (Section 9.5.1.7 SER and Section 9.5.1.8, SSER #3)

(a) Deleted per Amendment No. 169.

(b) Deleted per Amendment No. 169.

(c) The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the SNUPPS Final Safety Analysis Report for the facility through Revision 15, the Callaway site addendum through Revision 8, and as approved in the SER through Supplement 4, subject to provision d below.

(d) The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(e) Deleted (see Amendment No. 30, January 13, 1988)

(6) Qualification of Personnel (Section 13.1.2, SSER #3, Section 18, SSER #1)

Deleted per Amendment No. 169.

(7) NUREG-0737 Conditions (Section 22, SER)

Deleted per Amendment No. 169.

(8) Post-Fuel-Loading Initial Test Program (Section 14, SER)

Deleted per Amendment No. 169.

(9) Inservice Inspection Program (Sections 5.2.4 and 6.6, SER)

Deleted per Amendment No. 169.

(10) Emergency Planning

Deleted per Amendment No. 169.

Amendment 169

(11) Steam Generator Tube Rupture (Section 15.4.4, SSER #3)

Deleted per Amendment No. 169.

(12) Low Temperature Overpressure Protection (Section 15, SSER #3)

Deleted per Amendment No. 169.

(13) LOCA Reanalysis (Section 15, SSER #3)

Deleted per Amendment No. 169.

(14) Generic Letter 83-28

Deleted per Amendment No. 169.

(15) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available, pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

(16) Additional Conditions

The Additional Conditions contained in Appendix C, as revised through Amendment No. 180, are hereby incorporated into this

Revised by letter dated June 27, 2007

license. UE shall operate the facility in accordance with the Additional Conditions.

D. An Exemption from certain requirements of Appendix J to 10 CFR Part 50, are described in the October 9, 1984 staff letter. This exemption is authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, this exemption is hereby granted pursuant to 10 CFR 50.12. With the granting of this exemption the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. UE shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 10 CFR 73.21, are entitled: "Callaway Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan, Revision 0" submitted by letter dated October 20, 2004, as supplemented by the letter May 11, 2006.

F. Deleted per Amendment No. 169.

G. UE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at Midnight on October 18, 2024.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY H. R. DENTON

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Attachments/Appendices:

1. Attachment 1 (Deleted per Amendment No. 169)
2. Attachment 2 (Deleted per Amendment No. 169)
3. Appendix A - Technical Specifications (NUREG-1058, Revision 1)
4. Appendix B - Environmental Protection Plan
5. Appendix C – Additional Conditions

Date of Issuance: October 18, 1984

Revised by letter dated June 27, 2007

Technical Specification Example

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Three pressurizer safety valves shall be OPERABLE with lift settings ≥ 2460 psig and ≤ 2510 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > 290°F.

----------NOTE----------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 54 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1	Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two or more pressurizer safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE 4 with any RCS cold leg temperatures ≤ 290°F.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

Technical Specification Basis Example

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ 290°F, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

APPLICABLE ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require SAFETY safety valve actuation assume operation of three pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of three safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power,

b. Loss of reactor coolant flow,

c. Loss of external electrical load,

d. Loss of normal feedwater,

e. Loss of all AC power to station auxiliaries, and

f. Locked rotor.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, d, and e (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The three pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of three valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included,

BASES

APPLICABILITY (continued)

although the listed accidents may not require the safety valves for protection.

The LCO is not applicable in MODE 4 when any RCS cold leg temperatures are ≤ 290°F or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 54 hour exception is based on 18 hour outage time for each of the three valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two or more pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ 290°F within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below 290°F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by three pressurizer safety valves.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. FSAR, Chapter 15.
3. WCAP-7769, Rev. 1, June 1972.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants.

Classification of Plant Conditions from Trojan UFSAR

15.0.1 CLASSIFICATION OF PLANT CONDITIONS

The American Nuclear Society (ANS) classification of plant conditions divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are discussed in Section 15.0.1 and listed as follows:

(1) Condition I: Normal operation and operational transients.
(2) Condition II: Faults of moderate frequency.
(3) Condition III: Infrequent faults.
(4) Condition IV: Limiting faults.

The basic principle applied in relating design requirements to each of the conditions is that the most frequent occurrences must yield little or no radiological risk to the public, and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed to the extent allowed by considerations, such as the single failure criterion, in fulfilling this principle.

In the evaluation of the radiological consequences associated with initiation of a spectrum of accident conditions, numerous assumptions must be postulated. In many instances these assumptions are a product of extremely conservative judgments. This is due to the fact that many physical phenomena, in particular fission product transport under accident conditions, are presently not understood to the extent that accurate predictions can be made. Therefore, the set of assumptions postulated would predominantly determine the accident classification.

15.0.1.1 Condition I – Normal Operation and Operational Transients

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Conditions I operation.

A typical list of Condition I events is listed below:

(1) Steady-state and shutdown operations:
    (a) Power operation (≅5 to 100 percent of full power).
    (b) Startup up (or standby) (critical, 0 to 5 percent of full power).
    (c) Hot shutdown (subcritical, RHR system isolated).
    (d) Cold shutdown (subcritical, RHR System in operation).
    (e) Refueling.
(2) Operation with permissible deviations which may occur during continued operation as permitted by the plant technical specifications must be considered in conjunction with other operational modes. These include:
    (a) Operation with components or systems out of service.
    (b) Leakage from fuel with cladding defects.
    (c) Activity in the reactor coolant.
        1) Fission products.
        2) Corrosion products.
        3) Tritium
    (d) Operation with steam generator leaks up to the maximum allowed by Technical Specifications.
(3) Operational transients:
    (a) Plant heatup and cooldown (up to 100°F/hr for the RCS; 200°F/hr for the pressurizer).
    (b) Step load changes (up to ±10 percent).
    (c) Ramp load changes (up to 5 percent/min).
    (d) Load rejection up to and including design load rejection transient.

15.0.1.2 Condition II – Faults of Moderate Frequency

These faults at worst result in the reactor shutdown with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV category.
For the purposes of this report the following faults have been grouped into this category:

(1) Uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low power startup condition.
(2) Uncontrolled RCCA bank widhrwal at power.
(3) RCCA misalignment (except single RCCA withdrawal at full power).
(4) Chemical and Volume Control System (CVCS) malfunction that results in a decrease in boron concentration in the reactor coolant.
(5) Partial loss of forced reactor coolant flow.
(6) Startup of an inactive reactor coolant loop.
(7) Loss of external electrical load and/or turbine trip.
(8) Loss of normal feedwater flow.
(9) Coincident loss of onsite and external (offsite) a-c power to the station.
(10) Excessive heat removal due to feedwater system malfunctions (decrease in feedwater temperature and/or increase in feedwater flow).
(11) Equipment malfunction or operator failure that results in increasing steam flow.
(12) Inadvertent opening of a pressurizer safety or relief valve.
(13) Inadvertent opening of a steam generator relief or safety valve.
(14) Inadvertent operation of Emergency Core Cooling System (ECCS) during power operation.

An evaluation of the reliability of the Reliability of the Reactor Protection system (RPS) actuation following initiation of Condition II events has been completed and is presented in Reference 1 for the relay protection logic. Standard reliability engineering techniques were used to assess likelihood of the trip failure due to random component failures. Common-mode failures were also qualitatively investigated. It was concluded from the evaluation that the likelihood of no trip following initiation of Condition II events is extremely small (2×10^{-7} derived for random component failures).
The solid-state protection system design has been evaluated by the same methods as used for the relay system and the same order of magnitude of reliability is provided.
Hence, because of the high reliability of the RPS no special provision is provided to be taken in the design to cope with the consequences of Condition II events without trip.

15.0.1.3Condition III – Infrequent Faults

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A condition III fault will not by itself generate a Condition IV fault or result in a consequential loss of function or the RCS or Containment barriers.
For the purposes of this report the following faults have been grouped into this category:

(1) Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates emergency core cooling
(2) Minor secondary system pipe break.
(3) Inadvertent loading and operation of a fuel assembly in an improper position.
(4) Complete loss of forced reactor coolant flow.
(5) Postulated radioactive releases due to liquid tank failures.
(6) Single RCCA withdrawal at full power.

15.0.1.4Condition IV – Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. These are the most drastic occurrences which must be designed against and represent

limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline valves of 10 CFR 100. A single condition IV fault does not cause a consequential loss of required function of system needed to cope with the fault including those of the ECCS and the Containment. For the purposes of this report the following faults have been classified in this category:

(1) Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the larges pipe in the RCS (LOCA).

(2) Major secondary system (steam or feedwater) pipe ruptures.

(3) Steam generator tube failure.

(4) RCP shaft seizure.

(5) Design bases fuel handling accidents in the Containment and spent fuel storage buildings.

(6) Spectrum of rod ejection accidents.

The analyses of thyroid and whole body doses, resulting from events leading to fission product release, are presented along with each accident discussion. The fission product inventories which form a bases for these calculations are presented in Chapter 11 and Section 15.0.11.

Regulatory Guide 1.29, Seismic Design Classification

U.S. NUCLEAR REGULATORY COMMISSION

March 2007
Revision 4

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.29

(Draft was issued as DG-1156, dated October 2006)

SEISMIC DESIGN CLASSIFICATION

A. INTRODUCTION

General Design Criterion (GDC) 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), requires that nuclear power plant structures, systems, and components (SSCs) important to safety must be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

Toward that end, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 establishes quality assurance requirements for the design, construction, and operation of nuclear power plant SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those SSCs.

In addition, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, requires that all nuclear power plants must be designed so that certain SSCs remain functional if the safe-shutdown earthquake ground motion (SSE) occurs.[1] These plant features are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) or 10 CFR 100.11.[2]

This guide describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in identifying and classifying those features of light-water-reactor (LWR) nuclear power plants that must be designed to withstand the effects of the SSE.

This regulatory guide relates to information collections that are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 100, which the Office of Management and Budget (OMB) approved under OMB control numbers 3150-0011 and 3150-0093, respectively. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

After reviewing a number of applications for construction permits and operating licenses for boiling- and pressurized-water nuclear power plants, the NRC staff developed a seismic design classification system for identifying those plant features that must be designed to withstand the effects of the SSE. In so doing, the staff designated as Seismic Category I those SSCs that must be designed to remain functional if the SSE occurs.

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1 Appendix S to 10 CFR Part 50 applies to applicants for a design certification or combined license pursuant to 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," or a construction permit or operating license pursuant to 10 CFR Part 50 on or after January 10, 1997. However, the earthquake engineering criteria in Section VI of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria" (Ref. 2), continue to apply to operating license applicants or holders whose construction permit was issued before January 10, 1997.

2 Dose values set forth in 10 CFR Part 100, "Reactor Site Criteria" (Ref. 2), continue to apply to operating license applicants or holders whose construction permits were issued before January 10, 1997. However, application of 10 CFR 50.67, "Accident source term," with the alternative source terms identified in the latest edition of Regulatory Guide 1.183,"Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" (Ref. 3), is a voluntary option to meet the new positions in this regulatory guidance.

C. REGULATORY POSITION

1. The following SSCs of a nuclear power plant, including their foundations and supports, are designated as Seismic Category I and must be designed to withstand the effects of the SSE and remain functional. The titles and functions of these Seismic Category I SSCs for LWR designs are based on existing technology from prior applications. Certain SSCs previously considered Seismic Category I may no longer have a safety-related function requiring Seismic Category I classification, and certain passive SSCs in new LWR designs may be titled differently. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 shall apply to all activities affecting the safety-related functions of these SSCs:

   a. the reactor coolant pressure boundary

   b. the reactor core and reactor vessel internals

   c. systems[3] or portions thereof that are required for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system)

   d. systems[2] or portions thereof that are required for (1) reactor shutdown, (2) residual heat removal, or (3) cooling the spent fuel storage pool

   e. those portions of the steam systems of boiling-water reactors extending from the outermost containment isolation valve up to but ***not*** including the turbine stop valve, and connected piping of a nominal size of 6.35 cm (2.5 inches) or larger, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation (the turbine stop valve should be designed to withstand the SSE and maintain its integrity)

   f. those portions of the steam and feedwater systems of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping of a nominal size of 6.35 cm (2.5 inches) or larger, up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation

   g. cooling water, component cooling, and auxiliary feedwater systems[2] or portions thereof, including the intake structures, that are required for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, (4) residual heat removal from the reactor, or (5) spent fuel storage pool cooling

   h. cooling water and seal water systems[2] or portions thereof that are required for functioning of reactor coolant system components important to safety, such as reactor coolant pumps

   i. systems[2] or portions thereof that are required to supply fuel for emergency equipment

   j. all electrical and mechanical devices and circuitry between the process and the input terminals of the actuator systems involved in generating signals that initiate protective action

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[3] The system boundary includes those portions of the system required to accomplish the specified safety function and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure when the safety function is required.

k. systems[2] or portions thereof that are required for (1) monitoring and (2) actuating systems[4] important to safety

l. the spent fuel storage pool structure, including the fuel racks

m. the reactivity control systems (e.g., control rods, control rod drives, and boron injection system)

n. the control room, including its associated equipment and all equipment needed to maintain the control room within safe habitability limits for personnel and safe environmental limits for vital equipment

o. primary and secondary reactor containment

p. systems,[2] other than radioactive waste management systems,[5] not covered by items 1.a through 1.o above that contain or may contain radioactive material and of which postulated failure would result in conservatively calculated potential offsite doses [using meteorology as recommended in the latest editions of Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors" (Ref. 6), Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors" (Ref. 7), and Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" (Ref. 3)] that are more than 0.005 Sievert (0.5 rem) to the whole body or its equivalent to any part of the body or total effective dose equivalent (TEDE), as applicable

q. the Class 1E electrical systems, including the auxiliary systems for the onsite electric power supplies, that provide the emergency electric power needed for functioning of plant features included in items 1.a through 1.p above

2. Those portions of SSCs of which continued function is not required but of which failure could reduce the functioning of any plant feature included in items 1.a through 1.q above to an unacceptable safety level or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure.[6]

3. At the interface between Seismic Category I and non-Seismic Category I SSCs, the Seismic Category I dynamic analysis requirements should be extended to either the first anchor point in the non-seismic system or a sufficient distance into the non-Seismic Category I system so that the Seismic Category I analysis remains valid.

4. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of those portions of SSCs covered under Regulatory Positions 2 and 3 above.

5. Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants" (Ref. 8), provides guidance used to establish the design requirements for portions of fire protection SSCs to meet the requirements of GDC 2, as they relate to designing those SSCs to withstand the effects of the SSE.

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[4] See the latest edition of Regulatory Guide 1.151, "Instrument Sensing Lines" (Ref. 4).

[5] See the latest edition of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. 5).

[6] Wherever practical, structures and equipment of which failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. No backfitting is intended or approved in connection with its issuance.

Except in those cases in which an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the NRC staff will use the methods described in this guide to evaluate (1) submittals in connection with applications for construction permits, standard plant design certifications, operating licenses, early site permits, and combined licenses, and (2) submittals from operating reactor licensees who voluntarily propose to initiate system modifications if there is a clear nexus between the proposed modifications and the subject for which guidance is provided herein.

REGULATORY ANALYSIS / BACKFIT ANALYSIS

The regulatory analysis and backfit analysis for this regulatory guide are available in Draft Regulatory Guide DG-1156, "Seismic Design Classification" (Ref. 9). The NRC issued DG-1156 in October 2006 to solicit public comment on the draft of this Revision 4 of Regulatory Guide 1.29.

REFERENCES

1. *U.S. Code of Federal Regulations*, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.[7]

2. *U.S. Code of Federal Regulations*, Title 10, Part 100, , "Reactor Site Criteria," U.S. Nuclear Regulatory Commission, Washington, DC.[7]

3. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington, DC.[8]

4. Regulatory Guide 1.51, "Instrument Sensing Lines," U.S. Nuclear Regulatory Commission, Washington, DC.[8]

5. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[8]

6. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors," U.S. Nuclear Regulatory Commission, Washington, DC.[8]

7. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, Washington, DC.[8]

8. Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.[8]

9. Draft Regulatory Guide DG-1156, "Seismic Design Classification," U.S. Nuclear Regulatory Commission, Washington, DC, October 2006.[9]

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[7] All NRC regulations listed herein are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/cfr. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; email PDR@nrc.gov.

[8] All regulatory guides listed herein were published by the U.S. Nuclear Regulatory Commission or its predecessor, the U.S. Atomic Energy Commission. Most are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/. Single copies of regulatory guides may also be obtained free of charge by writing the Reproduction and Distribution Services Section, ADM, USNRC, Washington, DC 20555-0001, by fax to (301) 415-2289, or by email to DISTRIBUTION@nrc.gov. Active guides may also be purchased from the National Technical Information Service (NTIS). Details may be obtained by contacting NTIS at 5285 Port Royal Road, Springfield, Virginia 22161, online at http://www.ntis.gov, by telephone at (800) 553-NTIS (6847) or (703) 605-6000, or by fax to (703) 605-6900. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

[9] Draft Regulatory Guide DG-1156 is available electronically under Accession #ML062540294 in the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4209 by fax at (301) 415-3548, and by email to PDR@nrc.gov.

10 CFR 50, Appendix A

Definitions and Explanations

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) 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 functions.[2]

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

Criteria

I. Overall Requirements

Criterion 1--Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2--Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3--Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

10 CFR 100, Appendix A

I. Purpose

General Design Criterion 2 of Appendix A to part 50 of this chapter requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. It is the purpose of these criteria to set forth the principal seismic and geologic considerations which guide the Commission in its evaluation of the suitability of proposed sites for nuclear power plants and the suitability of the plant design bases established in consideration of the seismic and geologic characteristics of the proposed sites.

These criteria are based on the limited geophysical and geological information available to date concerning faults and earthquake occurrence and effect. They will be revised as necessary when more complete information becomes available.

II. Scope

These criteria, which apply to nuclear power plants, describe the nature of the investigations required to obtain the geologic and seismic data necessary to determine site suitability and provide reasonable assurance that a nuclear power plant can be constructed and operated at a proposed site without undue risk to the health and safety of the public. They describe procedures for determining the quantitative vibratory ground motion design basis at a site due to earthquakes and describe information needed to determine whether and to what extent a nuclear power plant need be designed to withstand the effects of surface faulting. Other geologic and seismic factors required to be taken into account in the siting and design of nuclear power plants are identified.

The investigations described in this appendix are within the scope of investigations permitted by § 50.10(c)(1) of this chapter.

Each applicant for a construction permit shall investigate all seismic and geologic factors that may affect the design and operation of the proposed nuclear power plant irrespective of whether such factors are explicitly included in these criteria. Additional investigations and/or more conservative determinations than those included in these criteria may be required for sites located in areas having complex geology or in areas of high seismicity. If an applicant believes that the particular seismology and geology of a site indicate that some of these criteria, or portions thereof, need not be satisfied, the specific sections of these criteria should be identified in the license application, and supporting data to justify clearly such departures should be presented.

These criteria do not address investigations of volcanic phenomena required for sites located in areas of volcanic activity. Investigations of the volcanic aspects of such sites will be determined on a case-by-case basis.

III. Definitions

As used in these criteria:

(a) The *magnitude* of an earthquake is a measure of the size of an earthquake and is related to the energy released in the form of seismic waves. *Magnitude* means the numerical value on a Richter scale.

(b) The *intensity* of an earthquake is a measure of its effects on man, on man-built structures, and on the earth's surface at a particular location. Intensity means the numerical value on the Modified Mercalli scale.

(c) The *Safe Shutdown Earthquake*[1] is 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 reactor coolant pressure boundary,

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.

TR 3.3.5 Seismic Monitoring Instrumentation

3.3.5 Seismic Monitoring Instrumentation

TR 3.3.5 The seismic monitoring instrumentation listed in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: At all times.

ACTIONS

----------NOTES----------

1. TR 3.0.3 and TR 3.0.4 are not applicable.
2. Separate Condition entry is allowed for each seismic monitoring instrument.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required seismic monitoring instruments inoperable.	A.1 Restore instrument(s) to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Prepare and submit a report to the NRC in accordance with 10 CFR 50.4 outlining the cause(s) of the malfunction(s) and the plans for restoring the instrument(s) to OPERABLE status.	10 days

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more required seismic monitoring instruments actuated during a seismic event.	C.1 Restore instrumentation to OPERABLE status.	24 hours
	AND	
	C.2 Perform a CHANNEL CALIBRATION on actuated instrument(s).	24 hours
	AND	
	C.3 Analyze data retrieved from actuated instrumentation to determine the magnitude of the vibratory ground motion.	10 days
	AND	
	C.4 Prepare and submit a report to the NRC in accordance with 10 CFR 50.4 describing the magnitude and frequency spectrum of the seismic event and the resultant effect upon facility features important to safety.	10 days

TECHNICAL REQUIREMENT SURVEILLANCES

----------NOTE----------

Refer to Table 3.3.5-1 to determine which TRSs apply for each seismic monitoring instrument.

SURVEILLANCE		FREQUENCY
TRS 3.3.5.1	Perform CHANNEL CHECK.	31 days
TRS 3.3.5.2	Perform COT.	6 months
TRS 3.3.5.3	Perform CHANNEL CALIBRATION.	18 months

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENT AND SENSOR LOCATION	MEASUREMENT RANGE	REQUIRED NUMBER OF INSTRUMENTS	SURVEILLANCE REQUIREMENTS
1. Triaxial Time-History Recording Accelerographs			
a. ST 6336 - Containment Base Slab	0 - 2 g	1	TRS 3.3.5.1[(a)] TRS 3.3.5.2 TRS 3.3.5.3
b. ST 6336 - Containment Wall	0 - 2 g	1	TRS 3.3.5.1[(a)] TRS 3.3.5.2 TRS 3.3.5.3
c. ST 6336 - Fuel Building, Elev. 93'	0 - 2 g	1	TRS 3.3.5.1[(a)] TRS 3.3.5.2 TRS 3.3.5.3
d. ST 6336 - Cable Spreading Room	0 - 2 g	1	TRS 3.3.5.1[(a)] TRS 3.3.5.2 TRS 3.3.5.3
e. ST 6336 - Free Field	0 - 2 g	1	TRS 3.3.5.1[(a)] TRS 3.3.5.2 TRS 3.3.5.3
2. Triaxial Peak Accelerographs			
a. SR 6340 - Emergency Diesel Generator	0 - 2 g	1	TRS 3.3.5.3
b SR 6340 - CCW Heat Exchanger	0 - 2 g	1	TRS 3.3.5.3
c. SR 6340 - Top of Control Building	0 - 2 g	1	TRS 3.3.5.3
d. SR 6340 - Top of Fuel Building	0 - 2 g	1	TRS 3.3.5.3
e. SR 6340 - Top of Intake Structure	0 - 2 g	1	TRS 3.3.5.3
f. SR 6340 - Top of Containment	0 - 2 g	1	TRS 3.3.5.3
g. SR 6340 - Bottom of Containment	0 - 2 g	1	TRS 3.3.5.3
3. Triaxial Response - Spectrum Recorders			
a. SR 6341 - Containment Foundation	-	1[(b)]	TRS 3.3.5.1 TRS 3.3.5.2 TRS 3.3.5.3

1. (a) Except seismic trigger.

(b) With control room indication.