REGULATORY GUIDE 1.70
REVISION 3

STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

LWR EDITION

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OFFICE OF STANDARDS DEVELOPMENT
U. S. NUCLEAR REGULATORY COMMISSION

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LWR EDITION

142 290

TABLE OF CONTENTS

*Lines indicate changes from Revision 2, dated September 1975.

142 291

TABLE OF CONTENTS (Continued) Page

** Interface applies to standard designs only. See Appendix A.

142 292

TABLE OF CONTENTS (Continued)

142 293

TABLE OF CONTENTS (Continued) Page

TABLE OF CONTENTS (Continued Page

142 295

TABLE OF CONTENTS (Continued) Page

*** This section includes an NSSS or BOP interface. See Appendix A.

TABLE OF CONTENTS (Continued) Page

142 297

TABLE OF CONTENTS (Continued) Page

142 298

TABLE OF CONTENTS (Continued) Page

142 299

TABLE OF CONTENTS (Continued)

142 300

TABLE OF CONTENTS (Continued) Page

TABLE OF CONTENTS (Continued)

142 302

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued) Page

142 304

INTRODUCTION

Section 50.34 of 10 CFR Part 50 requires that each application for a construction permit for a nuclear reactor facility include a Preliminary Safety Analysis Report (PSAR) and that each application for a license to operate such a facility include a Final Safety Analysis Report (FSAR). Section 50.34 specifies in general terms the information to be supplied in these Safety Analysis Reports (SARs). Further information was provided in a "Guide to the Organization and Contents of Safety Analysis Reports" issued by the Atomic Energy Commission* on June 30, 1966.

In the course of reviewing applications for construction permits and operating licenses, the AEC Regulatory staff found that most SARs as initially submitted did not provide sufficient information to permit the staff to conclude its review, and it was necessary for the staff to make specific requests for additional information. These requests, which are available in the NRC Public Document Room in the Dockets for individual cases, are a source of additional guidance to applicants.

In 1970, the Commission instituted a series of Safety Guides to inform applicants of solutions to specific safety issues that were determined to be acceptable to the Regulatory staff and the Advisory Committee on Reactor Safeguards. In 1971, a new series of Information Guides was initiated to list needed information that is frequently omitted from applications.

In November 1971, the AEC Director of Regulation announced that the Regulatory staff would make a preliminary review of each application for a construction permit or an operating license to determine whether sufficient information is included. If it is clear that a responsible effort has not been made to provide the information needed by the staff for its review, the licensing review would not be started until the application is reasonably complete. The Director of Regulation also indicated that additional guidance would be issued shortly. Accordingly, in February 1972, the Commission distributed for information and comment a proposed "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." It provided a standard format for these reports and identified the principal information needed by the staff for its review. Numerous comments were received, and a revised document reflecting those comments and superseding both the February 1972 issue and the 1966 guide was issued in October 1972.

In December 1972, the Commission combined the Safety Guide and Information Guide Series to form a new series with an expanded scope. This new series, designated the Regulatory Guide Series, is intended to provide guidance to applicants for and holders of all specific licenses or permits

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*The Atomic Energy Commission was abolished by the Energy Reorganization Act of 1974, which also created the Nuclear Regulatory Commission and gave it the licensing and related regulatory function of the AEC.

issued by the Commission. The "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (Revision 1) issued in October 1972 was later made a part of the Regulatory Guide Series and designated Regulatory Guide 1.70.

As developments in the nuclear industry occurred and changes became necessary in the Commission's requirements for information on which to base its findings requisite to the issuance of a permit or license, interim revisions to specific sections of the Standard Format were issued. These interim revisions were issued in a subseries of regulatory guides bearing the designation 1.70.X. Regulatory Guides 1.70.1 through 1.70.38 were issued as the need arose to update portions of Revision 1 to the Standard Format. All the changes included in these guides were incorporated into Revision 2 to the Standard Format, which was issued in September 1975. Accordingly, Regulatory Guides 1.70.1 through 1.70.38 were withdrawn.

The need for many of the changes that appeared in Revision 2 became evident during the development of a series of standard review plans for the guidance of staff reviewers who perform the detailed safety review of applications to construct or operate nuclear power plants. The individual standard review plans were combined into a single document, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants--LWR Edition" (NUREG-75/087), dated September 1975. The primary purpose of the Standard Review Plan is to improve the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. The Standard Review Plan identifies the specific areas of review and the associated acceptance criteria to guide the staff when reviewing an SAR. Incorporated into some sections of the Standard Review Plan are Branch Technical Positions, which, although less formal than regulatory guides, provide acceptable methods for satisfying the acceptance criteria.

Changes were made in the numbering of some Standard Format sections in Revision 2 to provide consistency with the corresponding sections of the Standard Review Plans in order to increase the efficiency of the staff review. Revision 3 to the Standard Format incorporates changes made to reflect public comments on Revision 2; to improve the consistency of the Standard Format with existing sections of the Standard Review Plan; to track recent revisions to the Standard Review Plan; and to provide guidelines for identifying and a format for submitting nuclear steam supply system (NSSS) and balance-of-plant (BOP) interfaces for standard designs. The purpose of the interface guidelines (Appendix A to the Standard Format) is to provide that, in such instances when a standard design is referenced by an applicant, the necessary safety-related interfaces will be accounted for to ensure that systems, structures, and components within the standard design will perform their intended safety functions.

The principal purpose of the SAR is to inform the Commission of the nature of the plant, the plans for its use, and the safety evaluations that have been performed to evaluate whether the plant can be constructed and operated without undue risk to the health and safety of the public. The SAR is the principal document for the applicant to provide the information needed to understand the basis on which this conclusion has been reached; it is the principal document referenced in the Construction Permit or Operating License that describes the basis on which the permit or license is issued; and it is the basic document used by NRC inspectors to determine whether the facility is being constructed and operated within the licensed conditions. Therefore, the information contained in the SAR should be timely, accurate, complete, and organized in a format that provides easy access.

Purpose of Standard Format

The purpose of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (hereinafter "Standard Format") is to indicate the information to be provided in the SAR and to establish a uniform format for presenting the information. Use of this format will help ensure the completeness of the information provided, will assist the Commission's staff and others in locating the information, and will aid in shortening the time needed for the review process.

Applicability of Standard Format

This Standard Format applies specifically to SARs for light-water-cooled nuclear power reactors. Two additional editions of the Standard Format have been prepared, one for high-temperature gas-cooled reactors (HTGR Edition) and one for liquid metal fast breeder reactors (LMFBR Edition). Copies may be obtained on written request to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

Use of Standard Format

The Standard Format represents a format for SARs that is acceptable to the NRC staff. Conformance with the Standard Format, however, is not required. Safety Analysis Reports with different formats will be acceptable to the staff if they provide an adequate basis for the findings requisite to the issuance of a license or permit. However, because it may be more difficult to locate needed information, the staff review time for such reports may be longer, and there is a greater likehood that the staff may regard the report as incomplete.

Upon receipt of an application, the NRC staff will perform a preliminary review to determine if the SAR provides a reasonably complete presentation of the information that is needed to form a basis for the findings required before issuance of a permit or license in accordance with 10 CFR § 2.101. The Standard Format will be used by the staff as a guideline to identify the type of information needed unless there is good reason for not doing so. If the SAR does not provide a reasonably complete presentation of the necessary information, further review of the application will not be initiated until a reasonably complete presentation is provided. The information provided in the SAR should be up to date with respect to the state of technology for nuclear power plants and should take into account recent changes in the NRC regulations and guides and in industry codes and standards, results of recent developments in nuclear reactor safety, and experience in the construction and operation of nuclear power plants.

The Standard Format should be used for both Preliminary Safety Analysis Reports and Final Safety Analysis Reports; however, any specific item that applies only to the FSAR will be indicated in the text by adding (FSAR) at the end of the guidance for that item. An entire section that is applicable only to the FSAR will be indicated by including (FSAR) following the heading.

Style and Composition

The applicant should strive for clear, concise presentations of the information provided in the SAR. Confusing or ambiguous statements and unnecessarily verbose descriptions do not contribute to expeditious technical review. Claims of adequacy of designs or design methods should be supported by technical bases.

The SAR should follow the numbering system and headings of the Standard Format at least to the headings with three digits, e.g., 2.4.2 Floods.

Appendices to the SAR should be used to provide supplemental information not explicitly identified in the Standard Format. Examples of such information are (1) summaries of the manner in which the applicant has treated matters addressed in NRC Regulatory Guides or proposed regulations and (2) supplementary information regarding calculational methods or design approaches used by the applicant or its agents.

Duplication of information should be avoided. Similar or identical information may be requested in various sections of the Standard Format because it is relevant to more than one portion of the plant; however, this information should be presented in the principal section and appropriately referenced in the other applicable sections of the SAR. For example, where piping and instrumentation diagrams for the same system are requested in more than one section of the Standard Format, duplicate diagrams need not be submitted provided all the information requested in all sections is included on the diagrams and is appropriately identified and referenced.

The design information provided in the SAR should reflect the most advanced state of design at the time of submission. If certain information identified in the Standard Format is not yet available at the time of submission of a PSAR because the design has not progressed sufficiently at the time of writing, the PSAR should provide the criteria and bases being used to develop the required information, the concepts and alternatives under consideration, and the schedule for completion of the design and submission of the missing information. In general, the PSAR should describe the preliminary design of the plant in sufficient detail to enable a definitive evaluation by the staff as to whether the plant can be constructed and operated without undue risk to the health and safety of the public.

Changes from the criteria, design, and bases set forth in the PSAR, as well as any new criteria, designs, and bases, should be identified in the FSAR. The reasons for and safety significance of each change should be discussed. The FSAR should describe in detail the final design of the plant as constructed.

Where numerical values are stated, the number of significant figures given should reflect the accuracy or precision to which the number is known. Where possible, estimated limits of error or uncertainty should be given.

Abbreviations should be consistent throughout the SAR and should be consistent with generally accepted usage. Any abbreviations, symbols, or special terms unique to the proposed plant or not in general usage should be defined in each chapter of the SAR where they are used.

Drawings, maps, diagrams, sketches, and charts should be employed where the information can be presented more adequately or conveniently by such means. Due concern should be taken to ensure that all information presented in drawings is legible, symbols are defined, and drawings are not reduced to the extent that visual aids are necessary to interpret pertinent items of information presented in the drawings.

Reports or other documents that are referenced in the text of the SAR should be listed at the end of the section in which they are referenced. In cases where proprietary documents are referenced, a nonproprietary summary of the document should also be referenced. Material incorporated into the application by reference should be listed in Chapter 1 (see Section 1.6 of the Standard Format).

<u>Revisions</u>

Data and text should be updated or revised by replacing pages. "Pen and ink" or "cut and paste" changes should <u>not</u> be used.

The changed or revised portion on each page should be highlighted by a "change indicator" mark consisting of a bold vertical line drawn in the margin opposite the binding margin. The line should be the same length as the portion actually changed.

All pages submitted to update, revise, or add pages to the report should show the date of change and a change or amendment number. A guide

142 309

page listing the pages to be inserted and the pages to be removed should accompany the revised pages.

All statements on a revised page should be accurate as of the date of the submittals.

Special care should be made to ensure that the main sections of the report are revised to reflect any design changes reported in supplemental information, i.e., responses to NRC staff requests for information or responses to regulatory positions.

Physical Specifications

All material submitted as part of the Safety Analysis Report should conform to specific standards as to the physical dimensions of page size, quality of paper and inks, and number of pages, exhibits, and attachments. More specifically:

1. Paper Size (not to exceed)

   Text pages: 8-1/2 x 11 inches.

   Drawings and graphics: 8-1/2 x 11 inches preferred; however, a larger size is acceptable provided:

   a. the bound side does not exceed 11 inches except where required for legibility, and

   b. the finished copy when folded does not exceed 8-1/2 x 11 inches.

2. Paper Stock

   Weight or substance: 20 pound for printing on both sides.
   16 to 20 pound for printing on one side only.

   Composition: wood chemical sulphite (no groundwood) and a pH of 5.5.

   Color: white is preferred, but pastel colors are acceptable provided the combination of paper stock and ink is suitable for microfilming.

3. Ink

   Color sufficiently dense to record on microfilm or image-copying equipment.

4. Page Margins

   A margin of no less than one inch should be maintained on the top, bottom, and binding side of all pages.

142 310

5. Printing

Composition: text pages should be single spaced.

Type font and style: must be suitable for microfilming.

Reproduction: may be mechanically or photographically reproduced. Text pages should preferably be printed on two sides with the image printed head to head.

6. Binding

Pages should be punched for standard 3-hole loose-leaf binder.

7. Page Numbering

Pages should be numbered with the two digits corresponding to the chapter and first-level section numbers followed by a hyphen and a sequential number within the section, i.e., the third page in Section 4.1 of Chapter 4 should be numbered 4.1-3. Do not number the entire report sequentially. (Note that because of the small number of pages in many sections, this Standard Format is numbered sequentially within each chapter.)

142 311

1. INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

The first chapter of the SAR should present an introduction to the report and a general description of the plant. This chapter should enable the reader to obtain a basic understanding of the overall facility without having to refer to the subsequent chapters. Review of the detailed chapters that follow can then be accomplished with better perspective and with recognition of the relative safety importance of each individual item to the overall plant design.

1.1 Introduction

This section should present briefly the principal aspects of the overall application, including the type of license requested, the number of plant units, a brief description of the proposed location of the plant, the type of the nuclear steam supply system and its designer, the type of containment structure and its designer, the core thermal power levels, both rated and design,* and the corresponding net electrical output for each thermal power level, the scheduled completion date, and the anticipated commercial operation date for each unit.

1.2 General Plant Description

This section should include a summary description of the principal characteristics of the site and a concise description of the plant. The plant description should include a brief discussion of the principal design criteria, operating characteristics, and safety considerations for the nuclear steam supply system; the engineered safety features and emergency systems; the instrumentation, control, and electrical systems; the power conversion system; the fuel handling and storage systems; the cooling water and other auxiliary systems; and the radioactive waste management system. The general arrangement of major structures and equipment should be indicated by the use of plan and elevation drawings in sufficient number and detail to provide a reasonable understanding of the general layout of the plant. Those features of the plant likely to be of special interest because of their relationship to safety should be identified. Such items as unusual site characteristics, solutions to particularly difficult engineering problems, and significant extrapolations in technology represented by the design should be highlighted.

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* Rated power is defined as the power level at which the plant would be operated if licensed. Design power is defined as the highest power level that would be permitted by plant design and that is used in some safety evaluations.

1.3 Comparison Tables

1.3.1 Comparisons with Similar Facility Designs

This section should provide a summary of sufficient detail to identify the principal similarities to other nuclear power plants (preferably plants already designed, constructed, or operated) and principal differences from such plants. Such comparisons may be limited to those plants or portions of plants designed or built by the nuclear steam system supplier, the architect-engineer, or the applicant. This information should be provided in tabular form with cross-references to the sections of the SAR that fully describe the similarities and differences. This comparison should not be restricted to a comparison of the reactor design parameters, but should include all principal features of the plant such as the engineered safety features, the containment concept, the instrumentation and electrical systems, the radioactive waste management system, and other principal systems.

1.3.2 Comparison of Final and Preliminary Information (FSAR)

The FSAR should be complete without reliance on the PSAR. In an FSAR, tables should be provided to identify clearly all the significant changes that have been made in the plant since submittal of the PSAR. Each item should be cross-referenced to the section in the FSAR that describes the changes and the reasons for them.

1.4 Identification of Agents and Contractors

This section should identify the prime agents or contractors for the design, construction, and operation of the nuclear power plant. The principal consultants and outside service organizations (such as those providing audits of the quality assurance program) should be identified. The division of responsibility between the reactor designer, architect-engineer, constructor, and plant operator should be delineated.

1.5 Requirements for Further Technical Information

This section of the PSAR should identify, describe, and discuss those safety features or components for which further technical information is required in support of the issuance of a construction permit, but which has not been supplied in the PSAR. This section of the PSAR should:

1. Identify and distinguish between those technical information development programs that will be required to determine the adequacy of a new design and those that will be used to demonstrate the margin of conservatism of a proven design,

2. Describe the specific technical information that must be obtained to demonstrate acceptable resolution of the problems,

142 313

3. Describe the program in sufficient detail to show how the information will be obtained, or cross-reference those sections of the PSAR in which this information is provided,

4. Provide a schedule of completion of the program as related to the projected startup date of the proposed plant, and

5. Discuss the design alternatives or operational restrictions available in the event that the results of the program do not demonstrate acceptable resolution of the problems.

Reference may be made to topical program summary reports filed with the NRC; however, if such references are made, the applicability of each technical information development item to the applicant's plant should be discussed.

In the FSAR, this section should include a résumé of special technical information development programs undertaken to establish the final design and/or demonstrate the conservatism of the design and a discussion of any programs that will be conducted during operation in order to demonstrate the acceptability of contemplated future changes in design or modes of operation.

1.6 Material Incorporated by Reference

This section should provide a tabulation of all topical reports that are incorporated by reference as part of the application. In this context, "topical reports" are defined as reports that have been prepared by reactor manufacturers, architect-engineers, or other organizations and filed separately with the NRC in support of this application or of other applications or product lines. This tabulation should include, for each topical report, the title, the report number, the date submitted to the NRC (or AEC), and the sections of the SAR in which this report is referenced. For any topical reports that have been withheld from public disclosure pursuant to Section 2.790(b) of 10 CFR Part 2 as proprietary documents, nonproprietary summary descriptions of the general content of such reports should also be referenced. This section should also include a tabulation of any documents submitted to the Commission in other applications that are incorporated in whole or in part in this application by reference. If any information submitted in connection with other applications is incorporated by reference in this SAR, summaries of such information should be included in appropriate sections of this SAR.

Results of tests and analyses may be submitted as separate reports. In such cases, these reports should be referenced in this section and summarized in the appropriate section of the SAR.

142 314

1.7 Drawings and Other Detailed Information

1.7.1 Electrical, Instrumentation, and Control Drawings (FSAR)

The FSAR should include a list of proprietary and nonproprietary electrical, instrumentation, and control (EI&C) drawings, including drawing number, title, revision number, and date. The list should be revised as necessary to conform to drawing revisions. Three copies of all proprietary and nonproprietary EI&C drawings, including revisions as they are issued, should be provided separate from the FSAR but incorporated by reference in this section.

1.7.2 Piping and Instrumentation Diagrams

For each piping and instrumentation diagram (including revisions as issued) in the SAR, two large-scale copies (approximately 22 in. x 34 in.) should be provided separately but should be referenced in this section. The piping and instrumentation diagrams should contain grid coordinates and drawing cross-references.

1.7.3 Other Detailed Information

This section of the SAR should include a list of other specific data submitted in response to requests of the NRC staff, including card decks for computer codes, computer printouts, and detailed geologic, seismologic, and foundation engineering information. Three copies of each such item should be submitted separately but should be referenced in this section.

1.8 Conformance to NRC Regulatory Guides

The SAR should include a table indicating the extent to which the applicant intends to comply with all applicable NRC regulatory guides and the revision number of those guides. For each applicable regulatory guide, the table should identify those sections of the SAR to which the guide applies and should indicate any proposed exceptions to the regulatory position.

1.9 Standard Designs

1.9.1 Interfaces

For standard designs, this section of the SSAR should provide a listing of the NSSS-BOP safety-related interfaces and should identify the sections in the SSAR where descriptions of these interfaces are presented.

1.9.2 Exceptions

In this section of the SAR, the applicant should (1) clearly identify and describe any exceptions taken to the approved standard design in the referenced SSAR and (2) reference the appropriate section in the SAR where

142 315

the detailed description of the component, system, or structure and the justification for the exception may be found. Portions of the SAR that contain exceptions to the referenced SSAR should be clearly identified by means of delineators such as marginal notation or pages of different color.

142 316

2. SITE CHARACTERISTICS

This chapter of the SAR should provide information on the geological, seismological, hydrological, and meteorological characteristics of the site and vicinity, in conjunction with present and projected population distribution and land use and site activities and controls. The purpose is to indicate how these site characteristics have influenced plant design and operating criteria and to show the adequacy of the site characteristics from a safety viewpoint.

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification of Location. The location of each reactor at the site should be specified by latitude and longitude to the nearest second and by Universal Transverse Mercator Coordinates (Zone Number, Northing, and Easting, as found on USGS topographical maps) to the nearest 100 meters. The State and county or other political subdivision in which the site is located should be identified, as well as the location of the site with respect to prominent natural and man-made features such as rivers and lakes.

2.1.1.2 Site* Area Map. A map of the site area of suitable scale (with explanatory text as necessary) should be included. It should clearly show the following:

1. The plant property lines. The area of plant property in acres should be stated.

2. Location of the site boundary. If the site boundary lines are the same as the plant property lines, this should be stated.

3. The location and orientation of principal plant structures within the site area. Principal structures should be identified as to function (e.g., reactor building, auxiliary building, turbine building).

4. The location of any industrial, commercial, institutional, recreational, or residential structures within the site area.

5. The boundary lines of the plant exclusion area (as defined in 10 CFR Part 100). If these boundary lines are the same as the plant property lines, this should be stated. The minimum distance from each reactor to the exclusion area boundary should be shown and specified.

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*"Site" means the contiguous real estate on which nuclear facilities are located and for which one or more licensees has the legal right to control access by individuals and to restrict land use for purposes of limiting the potential doses from radiation or radioactive material during normal operation of the facilities.

6. A scale that will permit the measurement of distances with reasonable accuracy.

7. True north.

8. Highways, railways, and waterways that traverse or are adjacent to the site.

2.1.1.3 Boundaries for Establishing Effluent Release Limits. The site description should define the boundary lines of the restricted area (as defined in 10 CFR Part 20) and should describe how access to this area is controlled for radiation protection purposes, including how the applicant will be made aware of individuals entering the area and will control such access. If it is proposed that limits higher than those established by § 20.106(a) (and related as low as is reasonably achievable provisions) be set, the information required by § 20.106 should be submitted. The site map discussed above may be used to identify this area, or a separate map of the site may be used. Indicate the location of the boundary line with respect to the water's edge of nearby rivers and lakes. Distances from plant effluent release points to the boundary line should be clearly defined.

2.1.2 Exclusion Area Authority and Control

2.1.2.1 Authority. The application should include a specific description of the applicant's legal rights with respect to all areas that lie within the designated exclusion area. The description should establish, as required by paragraph 100.3(a) of Part 100, that the applicant has the authority to determine all activities, including exclusion and removal of personnel and property from the area. The status of mineral rights and easements within this area should be addressed.

If ownership of all land within the exclusion area has not been obtained by the applicant, those parcels of land not owned within the area should be clearly described by means of a scaled map of the exclusion area, and the status of proceedings to obtain ownership or the required authority over the land for the life of the plant should be specifically described. Minimum distance to and direction of exclusion area boundaries should be given for both present ownership and proposed ownership. If the exclusion area extends into a body of water, the application should specifically address the bases upon which it has been determined that the authority required by paragraph 100.3(a) of Part 100 is or will be held by the applicant.

2.1.2.2 Control of Activities Unrelated to Plant Operation. Any activities unrelated to plant operation which are to be permitted within the exclusion area (aside from transit through the area) should be described with respect to the nature of such activities, the number of persons engaged in them, and the specific locations within the exclusion area where such activities will be permitted. The application should

142 31

describe the limitations to be imposed on such activities and the procedure to be followed to ensure that the plant staff has general knowledge of the number and location of persons within the exclusion area engaged in such activities. An estimate should be provided of the time required to evacuate all such persons from the area in order that calculations can be made of radiation doses resulting from the accidents postulated in Chapter 15.

2.1.2.3 Arrangements for Traffic Control. Where the exclusion area is traversed by a highway, railway, or waterway, the application should describe the arrangements made or to be made to control traffic in the event of an emergency.

2.1.2.4 Abandonment or Relocation of Roads. If there are any public roads traversing the proposed exclusion area which, because of their location, will have to be abandoned or relocated, specific information should be provided regarding authority possessed under state laws to effect abandonment; the procedures that must be followed to achieve abandonment; the identity of the public authorities who will make the final determination; and the status of the proceedings completed to date to obtain abandonment. If a public hearing is required prior to abandonment, the type of hearing should be specified (e.g., legislative or adjudicatory). If the public road will be relocated rather than abandoned, specific information as described above should be provided with regard to the relocation and the status of obtaining any lands required for relocation.

2.1.3 Population Distribution

Population data presented should be based on the 1970 census data and, where available, more recent census data. The following information should be presented on population distribution.

2.1.3.1 Population Within 10 Miles. On a map of suitable scale that identifies places of significant population grouping such as cities and towns within a 10-mile radius, concentric circles should be drawn, with the reactor at the center point, at distances of 1, 2, 3, 4, 5, and 10 miles. The circles should be divided into 22-1/2-degree segments with each segment centered on one of the 16 compass points (e.g., true north, north-northeast, northeast). A table appropriately keyed to the map should provide the current residential population within each area of the map formed by the concentric circles and radial lines. The same table, or separate tables, should be used to provide the projected population within each area for (1) the expected first year of plant operation and (2) by census decade (e.g., 1990) through the projected plant life. The tables should provide population totals for each segment and annular ring, and a total for the 0 to 10 miles enclosed population. The basis for population projections should be described.

2.1.3.2 Population Between 10 and 50 Miles. A map of suitable scale and appropriately keyed tables should be used in the same manner as

142 319

described above to describe the population and its distribution at 10-mile intervals between the 10- and 50-mile radii from the reactor.

2.1.3.3 Transient Population. Seasonal and daily variations in population and population distribution resulting from land uses such as recreational or industrial should be generally described and appropriately keyed to the areas and population numbers contained on the maps and tables of paragraphs 2.1.3.1 and 2.1.3.2. If the plant is located in an area where significant population variations due to transient land use are expected, additional tables of population distribution should be provided to indicate peak seasonal and daily populations. The additional tables should cover projected as well as current populations.

2.1.3.4 Low Population Zone. The low population zone (as defined in 10 CFR Part 100) should be specified and the basis for its selection discussed. A scaled map of the zone should be provided to illustrate topographic features; highways, railways, waterways, and any other transportation routes that may be used for evacuation purposes; and the location of all facilities and institutions such as schools, hospitals, prisons, beaches, and parks. Facilities and institutions beyond the low population zone which, because of their nature, may require special consideration when evaluating emergency plans, should be identified out to a distance of five miles. A table of population distribution within the low population zone should provide estimates of peak daily, as well as seasonal transient, population within the zone, including estimates of transient population in the facilities and institutions identified.

2.1.3.5 Population Center. The nearest population center (as defined in 10 CFR Part 100) should be identified and its population and its direction and distance from the reactor specified. The distance from the reactor to the nearest boundary of the population center (not necessarily the political boundary) should be related to the low population zone radius to demonstrate compliance with Part 100 guidelines. The bases for the boundary selected should be provided. Indicate the extent to which transient population has been considered in establishing the population center. In addition to specifying the distance to the nearest boundary of a population center, discuss the present and projected population distribution and population density within and adjacent to local population groupings.

2.1.3.6 Population Density. The cumulative resident population projected for the year of initial plant operation should be plotted to a distance of at least 30 miles and compared with a cumulative population resulting from a uniform population density of 500 people/sq. mile in all directions from the plant. Similar information should be provided for the end of plant life but compared with a cumulative population resulting from a uniform population density of 1000 people/sq. mile.

142 320

2.2 Nearby Industrial, Transportation, and Military Facilities

The purpose of this section is to establish whether the effects of potential accidents in the vicinity* of the site from present and projected industrial, transportation, and military installations and operations should be used as design basis events for plant design and to establish the design parameters related to the accidents so selected.

2.2.1 Locations and Routes

Provide maps showing the location and distance from the nuclear plant of all significant manufacturing plants; chemical plants; refineries; storage facilities; mining and quarrying operations; military bases; missile sites; transportation routes (air, land, and water); transportation facilities (docks, anchorages, airports); oil and gas pipelines, drilling operations, and wells; and underground gas storage facilities. Show any other facilities that, because of the products manufactured, stored, or transported, may require consideration with respect to possible adverse effects on the plant. Also, show any military firing or bombing ranges and any nearby aircraft flight, holding, and landing patterns.

The maps should be clearly legible and of suitable scale to enable easy location of the facilities and routes in relation to the nuclear plant. All symbols and notations used to depict the location of the facilities and routes should be identified in legends or tables. Topographic features should be included on the maps in sufficient detail to adequately illustrate the information presented.

2.2.2 Descriptions

The descriptions of the nearby industrial, transportation, and military facilities identified in 2.2.1 should include the information indicated in the following sections.

2.2.2.1 Description of Facilities. A concise description of each facility, including its primary function and major products and the number of persons employed, should be provided in tabular form.

2.2.2.2 Description of Products and Materials. A description of the products and materials regularly manufactured, stored, used, or transported in the vicinity of the nuclear plant should be provided. Emphasis should be placed on the identification and description of any hazardous materials. Statistical data should be provided on the amounts involved, modes of transportation, frequency of shipment, and the maximum quantity of hazardous material likely to be processed, stored, or transported at any given time. The applicable toxicity limits should be provided for each hazardous material.

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*All facilities and activities within five miles of the nuclear plant should be considered. Facilities and activities at greater distances should be included as appropriate to their significance.

142 321

2.2.2.3 Pipelines. For pipelines, indicate the pipe size, pipe age, operating pressure, depth of burial, location and type of isolation valves, and the type of gas or liquid presently carried. Indicate whether the pipeline is used for gas storage at higher than normal pressure and discuss the possibility of the pipeline being used in the future to carry a different product than the one presently being carried (e.g., propane instead of natural gas).

2.2.2.4 Waterways. If the site is located adjacent to a navigable waterway, provide information on the location of the intake structure(s) in relation to the shipping channel, the depth of channel, the location of locks, the type of ships and barges using the waterway, and any nearby docks and anchorages.

2.2.2.5 Airports. For airports, provide information on length and orientation of runways, type of aircraft using the facility, the number of operations per year by aircraft type, and the flying patterns associated with the airport. Plans for future utilization of the airport, including possible construction of new runways, increased traffic, or utilization by larger aircraft, should be provided. In addition, statistics on aircraft accidents* should be provided for:

1. All airports within five miles of the nuclear plant,
2. Airports with projected operations greater than $500d^2$ movements per year within 10 miles,** and
3. Airports with projected operations greater than $1000d^2$ movements per year outside 10 miles.**

Provide equivalent information describing any other aircraft activities in the vicinity of the plant. These should include aviation routes, pilot training areas, and landing and approach paths to airports and military facilities.

2.2.2.6 Projections of Industrial Growth. For each of the above categories, provide projections of the growth of present activities and new types of activities in the vicinity of the nuclear plant that can be reasonably expected based on economic growth projections for the area.

2.2.3 Evaluation of Potential Accidents

On the basis of the information provided in Sections 2.2.1 and .2, the potential accidents to be considered as design basis events should be determined and the potential effects of these accidents on the nuclear

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*An analysis of the probability of an aircraft collision at the nuclear plant and the effects of the collisi[illegible] the safety-related components of the plant should be provided in Section 3.5.

**"d" is the distance in miles from the site.

142 322

plant should be identified in terms of design parameter (e.g., overpressure, missile energies) or physical phenomena (e.g., concentration of flammable or toxic cloud outside building structures).

2.2.3.1 Determination of Design Basis Events. Design basis events external to the nuclear plant are defined as those accidents that have a probability of occurrence on the order of about 10^{-7} per year or greater and have potential consequences serious enough to affect the safety of the plant to the extent that Part 100 guidelines could be exceeded. The determination of the probability of occurrence of potential accidents should be based on an analysis of the available statistical data on the frequency of occurrence for the type of accident under consideration and on the transportation accident rates for the mode of transportation used to carry the hazardous material. If the probability of such an accident is on the order of 10^{-7} per year or greater, the accident should be considered a design basis event, and a detailed analysis of the effects of the accident on the plant's safety-related structures and components should be provided. The accident categories discussed below should be considered in selecting design basis events.

1. Explosions. Accidents involving detonations of high explosives, munitions, chemicals, or liquid and gaseous fuels should be considered for facilities and activities in the vicinity of the plant where such materials are processed, stored, used, or transported in quantity. Attention should be given to potential accidental explosions that could produce a blast overpressure on the order of 1 psi or greater at the nuclear plant, using recognized quantity-distance relationships.* Missiles generated in the explosion should also be considered, and an analysis should be provided in Section 3.5.

2. Flammable Vapor Clouds (Delayed Ignition). Accidental releases of flammable liquids or vapors that result in the formation of unconfined vapor clouds should be considered. Assuming that no immediate explosion occurs, the extent of the cloud and the concentrations of gas that could reach the plant under "worst-case" meteorological conditions should be determined. An evaluation of the effects on the plant of detonation and deflagration of the vapor cloud should be provided. An analysis of the missiles generated as a result of the detonation should be provided in Section 3.5.

3. Toxic Chemicals. Accidents involving the release of toxic chemicals (e.g., chlorine) from onsite storage facilities and nearby mobile and stationary sources should be considered. If toxic chemicals are known or projected to be present onsite or in the vicinity of a nuclear plant or to be frequently transported in the vicinity of the plant, releases

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*One acceptable reference is the Department of the Army Technical Manual TM 5-1300, "Structures to Resist the Effects of Accidental Explosions," for sale by Superintendent of Documents, U.S. Government Printing Office, Washington, D.C. 20402.

142 323

of these chemicals should be evaluated. For each postulated event, a range of concentrations at the site should be determined for a spectrum of meteorological conditions. These toxic chemical concentrations should be used in evaluating control room habitability in Section 6.4.

4. Fires. Accidents leading to high heat fluxes or to smoke, and nonflammable gas- or chemical-bearing clouds from the release of materials as the consequence of fires in the vicinity of the plant should be considered. Fires in adjacent industrial and chemical plants and storage facilities and in oil and gas pipelines, brush and forest fires, and fires from transportation accidents should be evaluated as events that could lead to high heat fluxes or to the formation of such clouds. A spectrum of meteorological conditions should be included in the dispersal analysis when determining the concentrations of nonflammable material that could reach the site. These concentrations should be used in Section 6.4 to evaluate control room habitability and in Section 9.5 to evaluate the operability of diesels and other equipment.

5. Collisions with Intake Structure. For nuclear power plant sites located on navigable waterways, the evaluation should consider the probability and potential effects of impact on the plant cooling water intake structure and enclosed pumps by the various size, weight, and type of barges or ships that normally pass the site, including any explosions incident to the collision. This analysis should be used in Section 9.2.5 to determine whether an additional source of cooling water is required.

6. Liquid Spills. The accidental release of oil or liquids which may be corrosive, cryogenic, or coagulant should be considered to determine if the potential exists for such liquids to be drawn into the plant's intake structure and circulating water system or otherwise to affect the plant's safe operation.

2.2.3.2 Effects of Design Basis Events. Provide the analysis of the effects of the design basis accidents identified in Section 2.2.3.1 on the safety-related components of the nuclear plant and discuss the steps taken to mitigate the consequences of these accidents, including such things as the addition of engineered-safety-feature equipment and reinforcing of plant structures, as well as the provisions made to lessen the likelihood and severity of the accidents themselves.

2.3 Meteorology

This section should provide a meteorological description of the site and its surrounding areas. Sufficient data should be included to permit an independent evaluation by the staff.

2.3.1 Regional Climatology

2.3.1.1 General Climate. The general climate of the region should be described with respect to types of air masses, synoptic features (high-

142 324

and low-pressure systems and frontal systems), general airflow patterns (wind direction and speed), temperature and humidity, precipitation (rain, snow, and sleet), and relationships between synoptic-scale atmospheric processes and local (site) meteorological conditions. Provide references that indicate the climatic atlases and regional climatic summaries used.

2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases. Seasonal and annual frequencies of severe weather phenomena, including hurricanes, tornadoes and waterspouts, thunderstorms, lightning, hail, and high air pollution potential, should be provided. Provide the probable maximum annual frequency of occurrence and time duration of freezing rain (ice storms) and dust (sand) storms where applicable. Provide estimates of the weight of the 100-year return period snowpack and the weight of the 48-hour Probable Maximum Winter Precipitation for the site vicinity. Using the above estimates, provide the weight of snow and ice on the roof of each safety-related structure.

Provide the meteorological data used for evaluating the performance of the ultimate heat sink with respect to (1) maximum evaporation and drift loss and (2) minimum water cooling (see Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"). The period of record examined should be identified, and the bases and procedures used for selection of the critical meteorological data should be provided and justified.

Provide design basis tornado parameters, including translational speed, rotational speed, maximum pressure differential with its associated time interval (see guidance in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"), and 100-year return period "fastest mile of wind," including vertical distribution of velocity and appropriate gust factor.

Provide all other regional meteorological and air quality conditions used for design and operating basis considerations and their bases. References to SAR sections in which these conditions are used should be included.

2.3.2 Local Meteorology

2.3.2.1 Normal and Extreme Values of Meteorological Parameters. Provide monthly and annual summaries (based on both long-term data from nearby reasonably representative locations and shorter-term onsite data) of:

1. Monthly and annual wind roses using the wind speed classes provided in Regulatory Guide 1.23 (Safety Guide 23), "Onsite Meteorological Programs," and wind direction persistence summaries at all heights at which wind characteristics data are applicable or have been measured.

2. Monthly and annual air temperature and dewpoint temperature summaries, including averages, measured extremes, and diurnal range.

3. Monthly and annual extremes of atmospheric water vapor (absolute and relative) including averages, measured extremes, and diurnal range.

4. Monthly and annual summaries of precipitation, including averages and measured extremes, number of hours with precipitation, rainfall rate distribution, (i.e., maximum 1 hr, 2 hr, ..., 24 hr) and monthly precipitation wind roses with precipitation rate classes.

5. Monthly and annual summaries of fog (and smog), including expected values and extremes of frequency and duration.

6. Monthly and annual summaries of atmospheric stability defined by vertical temperature gradient or other well-documented parameters that have been substantiated by diffusion data.

7. Monthly mixing height data, including frequency and duration (persistence) of inversion conditions.

8. Hourly averages of wind speed and direction at all heights at which wind characteristics data are applicable or have been measured and hourly averages of atmospheric stability as defined by vertical temperature gradient or other well-documented parameters that have been substantiated by diffusion data. (These data should be presented as hour-by-hour data on magnetic tape or monthly and annual joint frequency distributions of wind speed and wind direction by atmospheric stability.)

This information should be fully documented and substantiated as to the validity of its representation of conditions at and near the site. References should be provided to the National Weather Service (NOAA) station summaries from nearby locations and to other meteorological data that were used to describe site characteristics.

2.3.2.2 Potential Influence of the Plant and Its Facilities on Local Meteorology. Discuss and provide an evaluation of the potential modification of the normal and extreme values of meteorological parameters described in Section 2.3.2.1 above as a result of the presence and operation of the plant (e.g., the influence of cooling towers or water impoundment features on meteorological conditions). Provide a map showing the detailed topographic features (as modified by the plant) within a 5-mile (3.1 km) radius of the plant. Also provide a smaller scale map showing topography within a 50-mile (80 km) radius of the plant as well as a plot of maximum elevation versus distance from the center of the plant in each of the sixteen 22-1/2-degree compass point sectors (centered on true north, north-northeast, northeast, etc.) radiating fr the plant to a distance of 50 miles (80 km).

2.3.2.3 Local Meteorological Conditions for Design and Operating Bases. Provide all local meteorological and air quality conditions used for design and operating basis considerations and their bases, except for those conditions referred to in Sections 2.3.4 and 2.3.5. References should be included to SAR sections in which these conditions are used.

2.3.3 Onsite Meteorological Measurements Program

The preoperational and operational programs for meteorological measurements at the site, including offsite satellite facilities, should be described. This description should include measurements made, locations and elevations of measurements, exposure of instruments, descriptions of instruments used, instrument performance specifications, calibration and maintenance procedures, data output and recording systems and locations, and data analysis procedures. Additional sources of meteorological data for consideration in the description of airflow trajectories from the site to a distance of 80 km should be similarly described in as much detail as possible, particularly measurements made, locations and elevations of measurements, exposure of instruments, descriptions of instruments used, and instrument performance specifications. These additional sources of meteorological data may include National Weather Service stations and other meteorological programs that are well maintained and well exposed (e.g., other nuclear facilities, university and private meteorological programs). Guidance on acceptable onsite meteorological programs is presented in Regulatory Guide 1.23.

Provide joint frequency distributions of wind speed and direction by atmospheric stability class (derived from currently acceptable parameters), based on appropriate meteorological measurement heights and data reporting periods, in the format described in Regulatory Guide 1.23. An hour-by-hour listing of hourly-averaged parameters should also be provided on magnetic tape.

For the PSAR, at least one annual cycle of onsite meteorological data should be provided at docketing. If adequate meteorological data are not available at docketing, the best available (onsite and offsite) data to describe atmospheric dispersion characteristics should be provided. Adequate onsite meteorological data must be provided prior to or with the scheduled response to the first set of staff requests for additional information.

For the FSAR, at least two consecutive annual cycles (and preferably three or more whole years), including the most recent one-year period, should be provided at docketing.

Evidence should be provided to show how well these data represent long-term conditions at the site.

2.3.4 Short-Term Diffusion Estimates

2.3.4.1 Objective. Provide conservative and realistic estimates of atmospheric diffusion (χ/Q) at the site boundary (exclusion area) and at the outer boundary of the low population zone for appropriate time periods up to 30 days after an accident.

2.3.4.2 Calculations. Diffusion estimates should be based on the most representative meteorological data. Onsite data alone should be used as soon as a one-year period of record is completed.

Provide hourly cumulative frequency distributions of relative concentrations (χ/Q), using onsite data at appropriate distances from the effluent release point(s), such as the minimum site boundary distance (exclusion area). The χ/Q values from each of these distributions that are exceeded 5% and 50% (median value) of the time should be reported. For the outer boundary of the low population zone, provide cumulative frequency of χ/Q estimates for (1) the 8-hour time period from 0 to 8 hours; (2) the 16-hour period from 8 to 24 hours; (3) the 3-day period from 1 to 4 days; and (4) the 26-day period from 4 to 30 days. Report the worst condition and the 5% and 50% probability level conditions. Guidance on appropriate diffusion models for estimating χ/Q values is presented in Regulatory Guides 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors."

Evidence should be provided to show how well these diffusion estimates represent conditions that would be estimated from anticipated long-term conditions at the site. The effects of topography on short-term diffusion estimates should be discussed.

2.3.5 Long-Term Diffusion Estimates

2.3.5.1 Objective. Provide realistic estimates of annual average atmospheric transport and diffusion characteristics to a distance of 50 miles (80.5 km) from the plant for annual average release limit calculations and man-rem estimates.

2.3.5.2 Calculations. Provide a detailed description of the model used to calculate realistic annual average χ/Q values. Discuss the accuracy and validity of the model, including the suitability of input parameters, source configuration, and topography. Provide the meteorological data summaries (onsite and regional) used as input to the models. Guidance on acceptable atmospheric transport and diffusion models is presented in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors."

Provide a calculation of the maximum annual average χ/Q at or beyond the site boundary utilizing appropriate meteorological data for each routine venting location. Estimates of annual average χ/Q values for 16

142 328

radial sectors to a distance of 50 miles (80.5 km) from the plant using appropriate meteorological data should be provided.

Evidence should be provided to show how well these estimates represent conditions that would be estimated from climatologically representative data.

2.4 Hydrologic Engineering

The following sections should contain sufficient information to allow an independent hydrologic engineering review to be made of all hydrologically related design bases, performance requirements, and bases for operation of structures, systems, and components important to safety, considering the following phenomena or conditions:

1. Runoff floods for streams, reservoirs, adjacent drainage areas, and site drainage, and flood waves resulting from dam failures induced by runoff floods,

2. Surges, seiches, and wave action,

3. Tsunami,

4. Nonrunoff-induced flood waves due to dam failures or landslides,

5. Blockage of cooling water sources by natural events,

6. Ice jam flooding,

7. Combinations of flood types,

8. Low water and/or drought effects (including setdown due to surges, seiches, or tsunami) on safety-related cooling water supplies and their dependability,

9. Channel diversions of safety-related cooling water sources,

10. Capacity requirements for safety-related cooling water sources, and

11. Dilution and dispersion of severe accidental releases to the hydrosphere relating to existing and potential future users of surface water and groundwater resources.

The level of analysis that should be presented may range from very conservative, based on simplifying assumptions, to detailed analytical estimates of each facet of the bases being studied. The former approach is suggested in evaluating phenomena that do not influence the selection of design bases or where the adoption of very conservative design bases does not adversely affect plant design.

142 329

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities. Describe the site and all safety-related elevations, structures, exterior accesses, equipment, and systems from the standpoint of hydrologic considerations. Provide a topographic map of the site that shows any proposed changes to natural drainage features.

2.4.1.2 Hydrosphere. Describe the location, size, shape, and other hydrologic characteristics of streams, lakes, shore regions, and groundwater environments influencing plant siting. Include a description of existing and proposed water control structures, both upstream and downstream, that may influence conditions at the site. For these structures, (1) tabulate contributing drainage areas, (2) describe types of structures all appurtenances, ownership, seismic design criteria, and spillway design criteria, and (3) provide elevation-area-storage relationships and short-term and long-term storage allocations for pertinent reservoirs. Provide a regional map showing major hydrologic features. List the owner, location, and rate of use of surface water users whose intakes could be adversely affected by accidental release of contaminants. Refer to Section 2.4.13.2 for the tabulation of groundwater users.

2.4.2 Floods

2.4.2.1 Flood History. Provide the date, level, peak discharge, and related information for major historical flood events in the site region. A "flood" is defined as any abnormally high water stage or overflow from a stream, floodway, lake, or coastal area that results in significantly detrimental effects. Include stream floods, surges, seiches, tsunami, dam failures, ice jams, floods induced by landslides, and similar events.

2.4.2.2 Flood Design Considerations. Discuss the general capability of safety-related facilities, systems, and equipment to withstand floods and flood waves. The design flood protection for safety-related components and structures of the plant should be based on the highest calculated flood water level elevations and flood wave effects (design basis flood) resulting from analyses of several different hypothetical causes. Any possible flood condition up to and including the highest and most critical flood level resulting from any of several different events should be considered as the basis for the design protection level for safety-related components and structures of the plant. The flood potential from streams, reservoirs, adjacent watersheds, and site drainage should be discussed. The probable maximum water level from a stream flood, surge, seiche, combination of surge and stream flood in estuarial areas, wave action, or tsunami (whichever is applicable and/or greatest) may cause the highest water level at safety-related facilities. Other possibilities are the flood level resulting from the most severe flood wave at the plant site caused by an upstream or downstream landslide, dam failure, or dam breaching resulting from a seismic or foundation disturbance. The effects of coincident wind-generated wave action should be superimposed on the applicable flood level. The assumed hypothetical conditions should be evaluated both statically and dynamically to determine the design flood protection level.

The topical information that should be included is generally outlined in Sections 2.4.3 through 2.4.6 of this guide, but the types of events considered and the controlling event should be summarized in this section. Indicate whether, and if so how, the regulatory positions of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," have been followed; if not followed, describe the specific alternative approaches used.

2.4.2.3 Effects of Local Intense Precipitation. Describe the effects of local probable maximum precipitation (see Section 2.4.3.1) on adjacent drainage areas and site drainage systems, including drainage from the roofs of structures. Tabulate rainfall intensities for the selected and critically arranged time increments, provide characteristics and descriptions of runoff models, and estimate the resulting water levels. Summarize the design criteria for site drainage facilities and provide analyses that demonstrate the capability of site drainage facilities to prevent flooding of safety-related facilities resulting from local probable maximum precipitation. Estimates of precipitation based on NOAA publications (formerly U.S. Weather Bureau) with the time distribution based on critical distributions such as those employed by the Corps of Engineers usually provide acceptable bases. Sufficient details of the site drainage system should be provided (1) to allow an independent review of rainfall and runoff effects on safety-related facilities, (2) to judge the adequacy of design criteria, and (3) to allow independent review of the potential for blockage of site drainage due to ice, debris, or similar material.

Provide a discussion of the effects of ice accumulation on site facilities where such accumulation could coincide with local probable maximum (winter) precipitation and cause flooding or other damage to safety-related facilities.

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

Indicate whether, and if so how, the guidance given in Appendix A of Regulatory Guide 1.59 has been followed; if not followed, describe the specific alternative approaches used. Summarize the locations and associated water levels for which PMF determinations have been made.

2.4.3.1 Probable Maximum Precipitation (PMP). The PMP is the theoretical precipitation over the applicable drainage area that would produce flood flows that have virtually no risk of being exceeded. These estimates usually involve detailed analyses of actual storms in the general region of the drainage basin under study and certain modifications and extrapolations of historical data to reflect more severe rainfall conditions than have actually been recorded, insofar as these are deemed "reasonably possible" to occur on the basis of hydrometeorological reasoning. Discuss considerations of storm configuration (orientation of areal distribution), maximized precipitation amounts (include a description of maximization procedures and/or studies available for the area such as reference to National Weather Service and Corps of Engineers determinations), time distributions, orographic effects, storm centering, seasonal effects, antecedent storm sequences, antecedent snowpack (depth, moisture content, areal distribution), and any snowmelt model. Present the selected maximized storm precipitation distribution (time and space).

2.4.3.2 Precipitation Losses. Describe the absorption capability of the basin, including consideration of initial losses, infiltration rates, and antecedent precipitation. Provide verification of these assumptions by reference to regional studies or by presenting detailed applicable local storm-runoff studies.

2.4.3.3 Runoff and Stream Course Models. Describe the hydrologic response characteristics of the watershed to precipitation (such as unit hydrographs), verification from historical floods or synthetic procedures, and the nonlinearity of the model at high rainfall rates. A description of subbasin drainage areas (including a map), their sizes, and topographic features of watersheds should be provided. Include a tabulation of all drainage areas. Discuss the stream course model and its ability to compute floods up to the severity of the PMF. Present any reservoir and channel routing assumptions and coefficients and their bases with appropriate discussion of initial conditions, outlet works (controlled and uncontrolled), and spillways (controlled and uncontrolled).

2.4.3.4 Probable Maximum Flood Flow. Present the controlling PMF runoff hydrograph at the plant site that would result from rainfall (and snowmelt if pertinent). The analysis should consider all appropriate positions and distributions of the probable maximum precipitation and the potential influence of existing and proposed upstream and downstream dams and river structures. Present analyses and conclusions concerning the ability of upstream dams lying within a practical sphere of influence to withstand PMF conditions combined with setup, waves, and runup from appropriate coincident winds (see Section 2.4.3.6). If failures are likely, show the flood hydrographs at the plant site resulting from the most critical combination of such dam failures, including induced domino-type failures of dams lying upstream of the plant site. When credit is taken for flood lowering at the plant site as a result of failure of any downstream dam during a PMF, support the conclusion that the downstream dam is reasonably certain to fail. Finally, provide the estimated PMF discharge hydrograph at the site and, when available, provide a similar hydrograph without upstream reservoir effects to allow an evaluation of reservoir effects and a regional comparison of the PMF estimate to be made.

2.4.3.5 Water Level Determinations. Describe the translation of the estimated peak PMF discharge to elevation using (when applicable)

142 332

cross-section and profile data, reconstitution of historical floods (with consideration of high water marks and discharge estimates), standard step methods, transient flow methods, roughness coefficients, bridge and other losses, verification, extrapolation of coefficients for the PMF, estimates of PMF water surface profiles, and flood outlines.

2.4.3.6 Coincident Wind Wave Activity. Discuss setup, significant (33-1/3%) and maximum (1%) wave heights, runup, and resultant static and dynamic effects of wave action on each safety-related facility from wind-generated activity that may occur coincidently with the peak PMF water level. Provide a map and analysis showing that the most critical fetch has been used to determine wave action.

2.4.4 Potential Dam Failures, Seismically Induced

Indicate whether, and if so how, the guidance given in Appendix A of Regulatory Guide 1.59 has been followed; if not followed, describe the specific alternative approaches used.

2.4.4.1 Dam Failure Permutations. Discuss the locations of dams (both upstream and downstream), potential modes of failure, and results of seismically induced dam failures that could cause the most critical conditions (floods or low water) with respect to the safety-related facilities for such an event (see Section 2.4.3.4). Consideration should be given to possible landslides, preseismic-event reservoir levels, and antecedent flood flows coincident with the flood peak (base flow). Present the determination of the peak flow rate at the site for the worst dam failure reasonably possible or combination of dam failures, and summarize all analyses to show that the presented condition is the worst permutation. Include descriptions of all coefficients and methods used and their bases. Also, consider the effects on plant safety of other potential concurrent events such as blockage of a stream, waterborne missiles, etc.

2.4.4.2 Unsteady Flow Analysis of Potential Dam Failures. In determining the effect of dam failures at the site (see Section 2.4.4.1), the analytical methods presented should be applicable to artificially large floods with appropriately acceptable coefficients and should also consider flood waves through reservoirs downstream of failures. Domino-type failures resulting from flood waves should be considered, where applicable. Discuss estimates of coincident flow (see Regulatory Guide 1.59) and other assumptions used to attenuate the dam-failure flood wave downstream. Discuss static and dynamic effects of the attenuated wave at the site.

2.4.4.3 Water Level at Plant Site. Describe the backwater, unsteady flow, or other computational method leading to the water elevation estimate (Section 2.4.4.1) for the most critical upstream dam failure or failures, and discuss its verification and reliability. Superimpose wind and wave conditions that may occur simultaneously in a manner similar to that described in Section 2.4.3.6.

2.4.5 Probable Maximum Surge and Seiche Flooding

2.4.5.1 Probable Maximum Winds and Associated Meteorological Parameters. This mechanism is defined as a hypothetical hurricane or other windstorm that might result from the most severe combinations of meteorological parameters that are considered reasonably possible in the region involved, with the hurricane or other type of windstorm moving along a critical path and at an optimum rate of movement. The determination of probable maximum meteorological winds should be presented in detail. This determination involves detailed analyses of actual historical storm events in the general region and certain modifications and extrapolations of data to reflect a more severe meteorological wind system than actually recorded, insofar as these are deemed "reasonably possible" to occur on the basis of meteorological reasoning. Where this has been done previously or on a generic basis (e.g., Atlantic and Gulf Coast Probable Maximum Hurricane characteristics reported in U. S. Weather Bureau memorandum HUR 7-97), reference to that work with a brief description will be sufficient. The probable maximum conditions are the most severe combinations of hydrometeorological parameters considered reasonably possible that would produce a surge or seiche that has virtually no risk of being exceeded. This hypothetical event is postulated along a critical path at an optimal rate of movement from correlations of storm parameters of record. Sufficient bases and information should be provided to ensure that the parameters presented are the most severe combination.

2.4.5.2 Surge and Seiche Water Levels. Discuss considerations of hurricanes, frontal (cyclonic) type windstorms, moving squall lines, and surge mechanisms that are possible and applicable to the site. Include the antecedent water level (the 10% exceedance high tide, including initial rise for coastal locations, or the 100-year recurrence interval high water for lakes), the determination of the controlling storm surge or seiche (include the parameters used in the analysis such as storm track, wind fields, fetch or direction of wind approach, bottom effects, and verification of historic events), a detailed description of the methods and models used, and the results of the computation of the probable maximum surge hydrograph (graphical presentation).

2.4.5.3 Wave Action. Discuss the wind-generated wave activity that can occur coincidently with a surge or seiche, or independently. Estimates of the wave period and the significant (33-1/3%) and maximum (1%) wave heights and elevations with the coincident water level hydrograph should be presented. Specific data should be presented on the largest breaking-wave height, setup, runup, and the effect of overtopping in relation to each safety-related facility. A discussion of the effects of the water levels on each affected safety-related facility and the protection to be provided against static and dynamic effects and splash should be included.

2.4.5.4 Resonance. Discuss the possibility of oscillations of waves at natural periodicity, such as lake reflection and harbor resonance phenomena, and any resulting effects at the site.

142 334

2.4.5.5 Protective Structures. Discuss the location of and design criteria for any special facilities for the protection of intake, effluent, and other safety-related facilities against surges, seiches, and wave action.

2.4.6 Probable Maximum Tsunami Flooding

For sites adjacent to coastal areas, discuss historical tsunami, either recorded or translated and inferred, that provide information for use in determining the probable maximum water levels and the geoseismic generating mechanisms available, with appropriate references to Section 2.5.

2.4.6.1 Probable Maximum Tsunami. This event is defined as the most severe tsunami at the site, which has virtually no risk of being exceeded. Consideration should be given to the most reasonably severe geoseismic activity possible (resulting from, for example, fractures, faults, landslides, volcanism) in determining the limiting tsunami-producing mechanism. The geoseismic investigations required to identify potential tsunami sources and mechanisms are similar to those necessary for the analysis of surface faulting and vibratory ground motions indicated for Section 2.5 and are summarized herein to define those locations and mechanisms that could produce the controlling maximum tsunami at the site (from both local and distant generating mechanisms). Such considerations as the orientation of the site relative to the earthquake epicenter or generating mechanism, shape of the coastline, offshore land areas, hydrography, and stability of the coastal area (proneness of sliding) should be considered in the analysis.

2.4.6.2 Historical Tsunami Record. Provide local and regional historical tsunami information.

2.4.6.3 Source Generator Characteristics. Provide detailed geoseismic descriptions of the controlling local and distant tsunami generators, including location, source dimensions, fault orientation, and maximum displacement.

2.4.6.4 Tsunami Analysis. Provide a complete description of the analysis procedure used to calculate tsunami height and period at the site. All models used in the analysis should be described in detail. The description should include the theoretical bases of the model, its verification, and the conservatism of all input parameters.

2.4.6.5 Tsunami Water Levels. Provide estimates of maximum and minimum (low water) tsunami heights from both distant and local generators. Describe the ambient water levels, including tides, sea level anomalies, and wind waves assumed coincident with the tsunami.

2.4.6.6 Hydrography and Harbor or Breakwater Influences on Tsunami. Present the routing of the controlling tsunami, including breaking wave

142 335

formation, bore formation, and any resonance effects (natural frequencies and successive wave effects) that result in the estimate of the maximum tsunami runup on each pertinent safety-related facility. This should include a discussion both of the analysis used to translate tsunami waves from offshore generator locations, or in deep water, to the site and of antecedent conditions. Provide, where possible, verification of the techniques and coefficients used by reconstituting tsunami of record.

2.4.6.7 Effects on Safety-Related Facilities. Discuss the effects of the controlling tsunami on safety-related facilities and discuss the design criteria for the tsunami protection to be provided.

2.4.7 Ice Effects

Describe potential icing effects and design criteria for protecting safety-related facilities from the most severe ice jam flood, wind-driven ice ridges, or other ice-produced effects and forces that are reasonably possible and could affect safety-related facilities with respect to adjacent streams, lakes, etc., for both high and low water levels. Include the location and proximity of such facilities to the ice-generating mechanisms. Describe the regional ice and ice jam formation history with respect to water bodies.

2.4.8 Cooling Water Canals and Reservoirs

Present the design bases for the capacity and the operating plan for safety-related cooling water canals and reservoirs (reference Section 2.4.11). Discuss and provide bases for protecting the canals and reservoirs against wind waves, flow velocities (including allowance for freeboard), and blockage and (where applicable) describe the ability to withstand a probable maximum flood, surge, etc.

Discuss the emergency storage evacuation of reservoirs (low-level outlet and emergency spillway). Describe verified runoff models (e.g., unit hydrographs), flood routing, spillway design, and outlet protection.

2.4.9 Channel Diversions

Discuss the potential for upstream diversion or rerouting of the source of cooling water (resulting from, for example, river cutoffs, ice jams, or subsidence) with respect to historical, topographical, and geologic evidence in the region. Present the history of flow diversions and realignments in the region. Discuss the potential for adversely affecting safety-related facilities or water supply, and describe available alternative safety-related cooling water sources in the event that diversions are possible.

142 336

2.4.10 Flooding Protection Requirements

Describe the static and dynamic consequences of all types of flooding on each pertinent safety-related facility. Present the design bases required to ensure that safety-related facilities will be capable of surviving all design flood conditions, and reference appropriate discussions in other sections of the SAR where the design bases are implemented. Describe various types of flood protection used and the emergency procedures to be implemented (where applicable).

2.4.11 Low Water Considerations

2.4.11.1 Low Flow in Streams. Estimate and provide the design basis for the probable minimum flow rate and level resulting from the most severe drought considered reasonably possible in the region, if such conditions could affect the ability of safety-related facilities, particularly the ultimate heat sink, to perform adequately. Include considerations of downstream dam failures (see Section 2.4.4). For non-safety-related water supplies, demonstrate that the supply will be adequate during a 100-year drought.

2.4.11.2 Low Water Resulting from Surges, Seiches, or Tsunami. Determine the surge-, seiche-, or tsunami-caused low water level that could occur from probable maximum meteorological or geoseismic events, if such level could affect the ability of safety-related features to function adequately. Include a description of the probable maximum meteorological event (its track, associated parameters, antecedent conditions) and the computed low water level, or a description of tsunami conditions applicable. Also consider, where applicable, ice formation or ice jams causing low flow since such conditions may affect the safety-related cooling water source.

2.4.11.3 Historical Low Water. Discuss historical low water flows and levels and their probabilities (unadjusted for historical controls and adjusted for both historical and future controls and uses) only when statistical methods are used to extrapolate flows and/or levels to probable minimum conditions.

2.4.11.4 Future Controls. Provide the estimated flow rate, durations, and levels for probable minimum flow conditions considering future uses, if such conditions could affect the ability of safety-related facilities to function adequately. Substantiate any provisions for flow augmentation for plant use.

2.4.11.5 Plant Requirements. Present the required minimum safety-related cooling water flow, the sump invert elevation and configuration, the minimum design operating level, pump submergence elevations (operating heads), and design bases for effluent submergence, mixing, and dispersion. Discuss the capability of cooling water pumps to supply sufficient water during periods of low water resulting from the 100-year drought. Refer to Sections 9.2.1, 9.2.5, and 10.4.5 where applicable. Identify or refer to institutional restraints on water use.

2.4.11.6 Heat Sink Dependability Requirements. Identify all sources of normal and emergency shutdown water supply and related retaining and conveyance systems.

Identify design bases used to compare minimum flow and level estimates with plant requirements and describe any available low water safety factors (see Sections 2.4.4 and 2.4.11). Describe (or refer to Section 9.2.5) the design bases for operation and normal or accidental shutdown and cooldown during (1) the most severe natural and site-related accident phenomena, (2) reasonable combinations of less severe phenomena, and (3) single failures of man-made structural components. In the PSAR, describe or refer to the criteria for protecting all structures related to the ultimate heat sink during the above events. In the FSAR, describe the design to implement the criteria. Identify the sources of water and related retaining and conveyance systems that will be designed for each of the above bases or situations.

Describe the ability to provide sufficient warning of impending low flow or low water levels to allow switching to alternative sources where necessary. Heat dissipation capacity and water losses (such as drift, seepage, and evaporation) should be identified and conservatively estimated. Indicate whether, and if so how, guidance given in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," has been followed; if not followed, describe the specific alternative approaches used.

Identify or refer to descriptions of any other uses of water drawn from the ultimate heat sink, such as fire water or system charging requirements. If interdependent water supply systems are used, such as an excavated reservoir within a cooling lake or tandem reservoirs, describe the ability of the principal portion of the system to survive the failure of the secondary portion. Provide the bases for and describe the measures to be taken (dredging or other maintenance) to prevent loss of reservoir capacity as a result of sedimentation.

2.4.12 Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents in Surface Waters

Describe the ability of the surface water environment to disperse, dilute, or concentrate accidental liquid releases of radioactive effluents as related to existing or potential future water users. Discuss the bases used to determine dilution factors, dispersion coefficients, flow velocities, travel times, sorption and pathways of liquid contaminants. The locations and users of surface waters should be included in Section 2.4.1.2, and the release points should be identified in Section 11.2.3.

142 338

2.4.13 Groundwater

All groundwater data should be presented in this section, in Section 2.5.4, or in both and should be appropriately cross-referenced. If the information is placed in both sections, the information in the two sections should be consistent.

2.4.13.1 Description and Onsite Use. Describe the regional and local groundwater aquifers, formations, sources, and sinks. Describe the type of groundwater use, wells, pumps, storage facilities, and flow requirements of the plant. If groundwater is to be used as a safety-related source of water, the design basis protection from natural and accident phenomena should be compared with Regulatory Guide 1.27 guidelines and an indication should be given as to whether, and if so how, the guidelines have been followed; if not followed, the specific alternative approaches used should be described. Bases and sources of data should be adequately described.

2.4.13.2 Sources. Describe present regional use and projected future use. Tabulate existing users (amounts, water levels and elevations, locations, and drawdown). Tabulate or illustrate the history of groundwater or piezometric level fluctuations beneath and in the vicinity of the site. Provide groundwater or piezometric contour maps of aquifers beneath and in the vicinity of the site to indicate flow directions and gradients; discuss the seasonal and long-term variations of these aquifers. Indicate the range of values and the method of determination for vertical and horizontal permeability and total and effective porosity (specific yield) for each relevant geologic formation beneath the site. Discuss the potential for reversibility of groundwater flow resulting from local areas of pumping for both plant and nonplant use. Describe the effects of present and projected groundwater use (wells) on gradients and groundwater or piezometric levels beneath the site. Note any potential groundwater recharge area such as lakes or outcrops within the influence of the plant.

2.4.13.3 Accident Effects. Provide a conservative analysis of a postulated accidental release of liquid radioactive material at the site. Evaluate (where applicable) the dispersion, ion-exchange, and dilution capability of the groundwater environment with respect to present and projected users. Identify potential pathways of contamination to nearby groundwater users and to springs, lakes, streams, etc. Determine groundwater and radionuclide (if necessary) travel time to the nearest downgradient groundwater user or surface body of water. Include all methods of calculation, data sources, models, and parameters or coefficients used such as dispersion coefficients, dispersivity, distribution (sorption) coefficients hydraulic gradients, and values of permeability, total and effective porosity, and bulk density along contaminant pathways.

2.4.13.4 Monitoring or Safeguard Requirements. Present and discuss plans, procedures, safeguards, and monitoring programs to be used to protect present and projected groundwater users.

2.4.13.5 Design Bases for Subsurface Hydrostatic Loading.

1. For plants not employing permanent dewatering systems, describe the design bases for groundwater-induced hydrostatic loadings on subsurface portions of safety-related structures, systems, and components. Discuss the development of these design bases. Where dewatering during construction is critical to the integrity of safety-related structures, describe the bases for subsurface hydrostatic loadings assumed during construction and the dewatering methods to be employed in achieving these loadings.

Where wells are proposed for safety-related purposes, discuss the hydrodynamic design bases for protection against seismically-induced pressure waves. These design bases should be consistent with the groundwater conditions described in Sections 2.4.13.2 and 2.5.4.6.

2. For plants employing permanent dewatering systems:

a. Provide a description of the proposed dewatering system, including drawings showing the proposed locations of affected structures, components, and features of the system. Provide information related to the hydrologic design of all system components. Where the dewatering system is important to safety, provide a discussion of its expected functional reliability. The discussion of the bases for reliability should include comparisons of proposed systems and components with the performance of existing and comparable systems and components for applications under site conditions similar to those proposed.

b. Provide estimates and their bases for soil and rock permeabilities, total porosity, effective porosity (specific yield), storage coefficient, and other related parameters used in the design of the dewatering system. If available, provide the results of monitoring pumping rates and flow patterns during dewatering for the construction excavation.

c. Provide analyses and their bases for estimates of groundwater flow rates in the various parts of the permanent dewatering system, the area of influence of drawdown, and the shapes [illegible]reatic surfaces to be expected during operation of the system.

d. Provide analyses, including their bases, to establish conservative estimates of the time available to mitigate the consequences of the system degradation that could cause groundwater levels to exceed design bases. Document the measures that will be taken to repair the system or to provide an alternative dewatering system that would become operational before the design basis groundwater level is exceeded.

e. Provide both the design basis and normal operation groundwater levels for safety-related structures, systems, and components. The design basis groundwater level is defined as the maximum groundwater level used in the design analysis for dynamic or static loading conditions (whichever is being considered) and may be in excess of the elevation

142 340

for which the underdrain system is designed for normal operation. This level should consider abnormal and rare events (such as an occurrence of the Safe Shutdown Earthquake (SSE), a failure of a circulating water system pipe, or a single failure within the system) that can cause failure or overloading of the permanent dewatering system.

f. Postulate a single failure of a critical active feature or component during any design basis event. Unless it can be documented that the potential consequences of the failure will not result in dose guidelines exceeding those in Regulatory Guides 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classification," either (1) document by pertinent analyses that groundwater level recovery times are sufficient to allow other forms of dewatering to be implemented before the design basis groundwater level is exceeded, discuss the measures to be implemented and equipment needed, and identify the amount of time required to accomplish each measure or (2) design all system components for all severe phenomena and events.

g. Where appropriate, document the bases that ensure the ability of the system to withstand various natural and accidental phenomena such as earthquakes, tornadoes, surges, floods, and a single failure of a component feature of the system (such as a failure of any cooling water pipe penetrating, or in close proximity to, the outside walls of safety-related buildings where the groundwater level is controlled by the system). An analysis of the consequences of pipe ruptures on the proposed underdrain system should be provided and should include consideration of postulated breaks in the circulating system pipes at, in, or near the dewatering system building either independently of, or as a result of the SSE.

h. State the maximum groundwater level the plant structures can tolerate under various significant loading conditions in the absence of the underdrain system.

i. Provide a description of the proposed groundwater level monitoring programs for dewatering during plant construction and for permanent dewatering during plant operation. Provide (1) the general arrangement in plans and profile with approximate elevation of piezometers and observation wells to be installed, (2) intended zone(s) of placement, (3) type(s) of piezometer (closed or open system), (4) screens and filter gradation descriptions, (5) drawings showing typical installations showing limits of filter and seals, (6) observation schedules (initial and time intervals for subsequent readings), (7) plans for evaluation of [illegible]ed data, and (8) plans for alarm devices to ensure sufficient time f[illegible]ation of corrective action.

Provide information regarding the outlet flow monitoring program. The information required includes (1) the general location and type of flow measurement device(s) and (2) the observation plan and alarm

142 341

procedure to identify unanticipated high or low flow in the system and the condition of the effluent.

k. For OL reviews, but only if not previously reviewed by the staff, provide (1) substantiation of assumed design bases using information gathered during dewatering for construction excavation and (2) all other details of the dewatering system design that implement design bases established during the CP review.

l. For OL reviews, provide a technical specification for periods when the dewatering system may be exposed to sources of water not considered in the design. An example of such a situation would be the excavation of surface seal material for repair of piping such that the underdrain would be exposed to direct surface runoff. In addition, where the permanent dewatering system is safety related, is not completely redundant, or is not designed for all design basis events, provide the bases for a technical specification with action levels, the remedial work required and the estimated time that it will take to accomplish the work, and the sources, types of equipment, and manpower required as well as the availability of the above under potentially adverse conditions.

m. Where wells are proposed for safety-related purposes, discuss the hydrodynamic design bases for protection against seismically-induced pressure waves. These design bases should be consistent with the groundwater conditions described in Section 2.4.13.2 and 2.5.4.6.

2.4.14 Technical Specification and Emergency Operation Requirements

Describe any emergency protective measures designed to minimize the impact of adverse hydrology-related events on safety-related facilities. Describe the manner in which these requirements will be incorporated into appropriate technical specifications and emergency procedures. Discuss the need for any technical specifications for plant shutdown to minimize the consequences of an accident resulting from hydrologic phenomena such as floods or the degradation of the ultimate heat sink. In the event emergency procedures are to be used to meet safety requirements associated with hydrologic events, identify the event, present appropriate water levels and lead times available, indicate what type of action would be taken, and discuss the time required to implement each procedure.

2.5 Geology, Seismology, and Geotechnical Engineering

This section of the SAR should provide information regarding the seismic and geologic characteristics of the site and the region surrounding the site. Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," gives the principal seismic and geologic considerations that guide the staff in its evaluation of the acceptability of sites and seismic design bases.

This section should include, but not necessarily be limited to, the information discussed below. It should be preceded by a summary that contains a synopsis of Sections 2.5.1 through 2.5.6. Include a brief description of the sites, the investigations performed, results of investigations, conclusions, and a statement as to who did the work.

2.5.1 Basic Geologic and Seismic Information

Basic geologic and seismic information is required throughout the following sections to provide a basis for evaluation. In some cases, this information is germane to more than one section. The information may be presented under this section, under the following sections, or as appendices to this section, provided adequate cross-references are made in the appropriate sections.

Information obtained from published reports, maps, private communications, or other sources should be referenced. Information from surveys, geophysical investigations, borings, trenches, or other investigations should be adequately documented by descriptions of techniques, graphic logs, photographs, laboratory results, identification of principal investigators, and other data necessary to assess the adequacy of the information.

2.5.1.1 Regional Geology. Discuss all geologic, seismic, and man-made hazards within the site region and relate them to the regional physiography, tectonic structures and tectonic provinces, geomorphology, stratigraphy, lithology, and geologic and structural history, and geochronology. The above information should be discussed, documented by appropriate references, and illustrated by a regional physiographic map, surface and subsurface geologic maps, isopach maps, regional gravity and magnetic maps, stratigraphic sections, tectonic and structure maps, fault maps, a site topographic map, a map showing areas of mineral and hydrocarbon extraction, boring logs, aerial photographs, and any maps needed to illustrate such hazards as subsidence, cavernous or karst terrain, irregular weathering conditions, and landslide potential.

The relationship between the regional and the site physiography should be discussed. A regional physiographic map showing the site location should be included. Identify and describe tectonic structures such as folds, faults, basins, and domes underlying the region surrounding the site, and include a discussion of their geologic history. A regional tectonic map showing the site location should be included and detailed discussions of the regional tectonic structures of significance to the site should be provided. The detailed analyses of faults to determine their capacity for generating ground motions at the site and to determine the potential for surface faulting should be included in Sections 2.5.2 and 2.5.3, respectively.

The lithologic, stratigraphic, and structural geologic conditions of the region surrounding the site should be described and related to

142 343

its geologic history. Provide geologic profiles showing the relationship of the regional and local geology to the site location. The geologic province within which the site is located and the relation to other geologic provinces should be indicated. Regional geologic maps indicating the site location and showing both surface and bedrock geology should also be included.

2.5.1.2 Site Geology. Material on site geology included in this section may be cross-referenced in Section 2.5.4. The site physiography and local land forms should be described and the relationship between the regional and site physiography should be discussed. A site topographic map showing the locations of the principal plant facilities should be included. Describe the configuration of the land forms and relate the history of geologic changes that have occurred. Areas that are significant to the site of actual or potential landsliding, surface or subsurface subsidence, uplift, or collapse resulting from natural features such as tectonic depression and cavernous or karst terrains should be evaluated.

The detailed lithologic and stratigraphic conditions of the site and the relationship to the regional stratigraphy should be described. The thicknesses, physical characteristics, origin, and degree of consolidation of each lithologic unit should also be described, including a local stratigraphic column. Furnish summary logs or borings and excavations such as trenches used in the geologic evaluation. Boring logs included in Section 2.5.4 may be referenced.

A detailed discussion of the structural geology in the vicinity of the site should be provided. Include in the discussion the relationship of site structure to regional tectonics, with particular attention to specific structural units of significance to the site such as folds, faults, synclines, anticlines, domes, and basins. Provide a large-scale structural geology map (1:24,000) of the site showing bedrock surface contours and including the locations of Seismic Category I structures. A large-scale geologic map (1:24,000) of the region within 5 miles of the site that shows surface geology and that includes the locations of major structures of the nuclear power plant, including all Seismic Category I structures, should also be furnished. Areas of bedrock outcrop from which geologic interpretation has been extrapolated should be distinguished from areas in which bedrock is not exposed at the surface. When the interpretation differs substantially from the published geologic literature on the area, the differences should be noted and documentation for the new conclusions presented.

The geologic history of the site should be discussed and related to the regional geologic history.

Include an evaluation from an engineering-geology standpoint of the local geologic features that affect the plant structures. Geologic conditions underlying all Seismic Category I structures, dams, dikes, and pipelines should be described in detail. The dynamic behavior of the

142 344

site during prior earthquakes should be described. Deformational zones such as shears, joints, fractures, and folds, or combinations of these features should be identified and evaluated relative to structural foundations. Describe and evaluate zones of alteration or irregular weathering profiles, zones of structural weakness, unrelieved residual stresses in bedrock, and all rocks or soils that might be unstable because of their mineralogy or unstable physical or chemical properties. The effects of man's activities in the area such as withdrawal or addition of subsurface fluids or mineral extraction at the site should be evaluated.

Site groundwater conditions should be described. Information included in Section 2.4.13 may be referenced in this section.

2.5.2 Vibratory Ground Motion

This section is directed toward establishing the seismic design basis for vibratory ground motion. The presentation should be aimed at (1) determining the Safe Shutdown Earthquake (SSE) and the Operating Basis Earthquake (OBE) for the site and (2) specifying the vibratory ground motion corresponding to each of these events. Determination of the SSE and the OBE should be based on the identification of tectonic provinces or active geologic structures with which earthquake activity in the region can be associated. The design vibratory ground motion for the SSE and OBE should then be determined by assessing the effects at the site of the SSE and OBE associated with the identified provinces or structures.

The presentation in the SAR should proceed from discussions of the regional seismicity, geologic structures, and tectonic activity to a determination of the relation between seismicity and geologic structures. Earthquake-generating potential of tectonic provinces and any active structures should be identified. Finally, the ground motion that would result at the site from the maximum potential earthquakes associated with each tectonic province or geologic structure should be assessed considering any site amplification effects. The results should be used to establish the vibratory ground motion design spectrum.

Information should be presented to describe how the design basis for vibratory ground motion (Safe Shutdown Earthquake) was determined. The following specific information and determinations should also be included, as needed, to clearly establish the design basis for vibratory ground motion. Information presented in other sections may be cross-referenced and need not be repeated.

2.5.2.1 Seismicity. A complete list of all historically reported earthquakes that could have reasonably affected the region surrounding the site should be provided. The listing should include all earthquakes of MM Intensity greater than IV or magnitude greater than 3.0 that have been reported in all tectonic provinces, any part of which is within 200 miles of the site. This account should be augmented by a regional-scale map showing all listed earthquake epicenters and, in areas of high seismicity, by a larger-scale map showing earthquake epicenters within 50 miles of the site. The following information describing each earthquake should be provided whenever it is available: epicenter coordinates, depth of focus, origin time, highest intensity, magnitude, seismic moment, source mechanism, source dimensions, source rise time, rupture velocity, total dislocation, fractional stress drop, any strong-motion recordings, and identification of references from which the specified information was obtained. In addition, any earthquake-induced geologic hazards (e.g., liquefaction, landsliding, landspreading, or lurching) that have been reported should be described completely, including the level of strong motion that induced failure and the properties of the materials involved.

2.5.2.2 Geologic Structures and Tectonic Activity. Identify the regional geologic structures and tectonic activity that are significant in determining regional earthquake potential. All tectonic provinces any part of which occurs within 200 miles of the site should be identified. The identification should include a description of those characteristics of geologic structure, tectonic history, present and past stress regimes, and seismicity that distinguish the various tectonic provinces and particular areas within those provinces where historical earthquakes have occurred. Alternative models of regional tectonic activity from available literature sources should be discussed. The discussion in this section should be augmented by a regional-scale map showing the tectonic provinces, earthquake epicenters, the locations of geologic structures and other features that characterize the provinces, and the locations of any capable faults.

2.5.2.3 Correlation of Earthquake Activity with Geologic Structures or Tectonic Provinces. Provide a correlation between epicenters or regions of highest intensity of historically reported earthquakes and geologic structures or tectonic provinces. Whenever an earthquake epicenter or concentration of earthquake epicenters can be reasonably correlated with geologic structures, the rationale for the association should be developed. This discussion should include identification of the methods used to locate the earthquake epicenters and an estimate of their accuracy and should provide a detailed account that compares and contrasts the geologic structure involved in the earthquake activity with other areas within the tectonic province. When an earthquake epicenter cannot be reasonably correlated with geologic structures, the epicenter should be discussed in relation to tectonic provinces. A subdivision of a tectonic province should be corroborated on the basis of evaluations that consider, but should not be limited to, detailed seismicity studies, tectonic flux measurements, contrasting structural fabric, differences in geologic history, and differences in stress regime.

2.5.2.4 Maximum Earthquake Potential. The largest earthquakes associated with each geologic structure or tectonic province should be identified. Where the earthquakes are associated with a geologic structure, the largest earthquake that could occur on that structure should be evaluated based on considerations such as the nature of faulting, fault length,

142 346

fault displacement, and earthquake history. Where the earthquakes are associated with a tectonic province, the largest historical earthquakes within the province should be identified and, whenever reasonable, the return period for the earthquakes should be determined. Isoseismal maps should also be presented for the earthquakes.

Ground motion at the site should be determined assuming seismic energy transmission effects are constant over the region and assuming the largest earthquake associated with each geologic structure or with each tectonic province occurs at the point of closest approach of that structure or province to the site. The set of conditions describing the occurrence of the potential earthquake that would produce the largest vibratory ground motion at the site should be defined. If different potential earthquakes would produce the maximum ground motion in different frequency bands, the conditions describing all such earthquakes should be specified. The description of the potential earthquake occurrences should include the maximum intensity or magnitude and the distance from the assumed location of the potential earthquake to the site.

2.5.2.5 Seismic Wave Transmission Characteristics of the Site. The following material properties should be determined for each stratum under the site: seismic compressional and shear velocities, bulk densities, soil properties and classification, shear modulus and its variation with strain level, and water table elevation and its variation. The methods used to determine these properties should be described. For each set of conditions describing the occurrence of the maximum potential earthquakes, determined in Section 2.5.2.4, the types of seismic waves producing the maximum ground motion and the significant frequencies at the site should be determined. For each set of conditions, an analysis should be performed to determine the effects of transmission in the site material for the identified seismic wave types in the significant frequency bands.

2.5.2.6 Safe Shutdown Earthquake. The acceleration at the ground surface, the effective frequency range, and the duration corresponding to each maximum potential earthquake should be determined. Where the earthquake has been associated with a geologic structure, the acceleration should be determined using a relation between acceleration, magnitude, or fault length, earthquake history and other geologic information, and the distance from the fault. Where the earthquake has been associated with a tectonic province, the acceleration should be determined using appropriate relations between acceleration, intensity, epicentral intensity, and distance. Available ground motion time histories from earthquakes of comparable magnitude, epicentral distance, and acceleration level should be presented. The spectral content from each maximum potential earthquake should be described based on consideration of the available ground motion time histories and regional characteristics of seismic wave transmission. The dominant frequency associated with the peak acceleration should be determined either from analysis of ground motion time histories or by inference from descriptions of earthquake phenomenology, damage reports, and regional characteristics of seismic wave transmission. Design

142 347

response spectra corresponding to the SSE should be defined and their conservatism assessed by comparing them to the ground motion expected from the potential earthquakes.

2.5.2.7 Operating Basis Earthquake. The vibratory ground motion for the Operating Basis Earthquake should be described and the probability of exceeding the OBE during the operating life of the plant should be determined.

2.5.3 Surface Faulting

Information should be provided to describe whether or not there exists a potential for surface faulting at the site. The following specific information and determinations should also be included to the extent necessary to clearly establish zones requiring detailed faulting investigation. Information presented in Section 2.5.1 may be cross-referenced and need not be repeated.

2.5.3.1 Geologic Conditions of the Site. The lithologic, stratigraphic, and structural geologic conditions of the site and the area surrounding the site, including its geologic history, should be described. Site and regional geologic maps and profiles illustrating the surface and bedrock geology, structure geology, topography, and the relationship of the safety-related foundations of the nuclear power plant to these features should be included.

2.5.3.2 Evidence of Fault Offset. Determine the geologic evidence of fault offset at or near the ground surface at or near the site. If faulting exists, it should be defined as to its attitudes, orientations, width of shear zone, amount and sense of movement, and age of movements. Any topographic photo linears and Environmental Resources Technology Satellite linears prepared as part of this study should be discussed. Site surface and subsurface investigations to determine the absence of faulting should be reported, including information on the detail and areal extent of the investigation.

2.5.3.3 Earthquakes Associated with Capable Faults. List all historically reported earthquakes that can be reasonably associated with faults, and part of which is within 5 miles of the site. A plot of earthquake epicenters superimposed on a map showing the local tectonic structures should be provided.

2.5.3.4 Investigation of Capable Faults. Identified faults, any part of which is within 5 miles of the site, should be investigated in sufficient detail and using geological and geophysical techniques of sufficient sensitivity to demonstrate the age of most recent movement on each. The type and extent of investigation varies from one geologic province to another and depends on site-specific conditions.

2.5.3.5 Correlation of Epicenters with Capable Faults. The structure and genetic relationship between site area faulting and regional tectonic

142 348

framework should be discussed. In regions of active tectonism, any detailed geologic and geophysical investigations conducted to demonstrate the structural relationships of site area faults with regional faults known to be seismically active should be discussed.

2.5.3.6 Description of Capable Faults. For capable faults more than 1,000 feet long, any part of which is within 5 miles of the site, determine for all offsets within the immediate site vicinity the length of the fault; the relationship to regional tectonic structures; the nature, amount, and geologic displacement along the fault; and the outer limits of the fault zone established by detailed faulting investigation.

2.5.3.7 Zone Requiring Detailed Faulting Investigation. Determine the zone requiring detailed faulting investigation as described in Appendix A to 10 CFR Part 100.

2.5.3.8 Results of Faulting Investigation. Where the site is located within a zone requiring detailed faulting investigation, details and the results of investigations should be provided to substantiate that there are no geologic hazards that could affect the safety-related facilities of the plant. The information may be in the form of boring logs, detailed geologic maps, geophysical data, maps and logs of trenches, remote sensing data, and seismic refraction and reflection data.

2.5.4 Stability of Subsurface Materials and Foundations

Information should be presented that thoroughly defines the conditions and engineering properties of both soil and/or rock supporting nuclear power plant foundations. The stability of the soils and rock under plant structures should be evaluated both for static and dynamic loading conditions (including an evaluation of the ability of these materials to perform their support function without incurring unexpected or excessive subsidence and settlement due to their long-term consolidation under load or to their response to natural phenomena). Both the operating and safe shutdown earthquakes should be used in the dynamic stability evaluation. An evaluation of site conditions and geologic features that may affect nuclear power plant structures or their foundations should be presented. Information presented in other sections should be cross-referenced rather than repeated.

2.5.4.1 Geologic Features. Describe geologic features, including the following:

1. Areas of actual or potential surface or subsurface subsidence, uplift, or collapse and the causes of these conditions,

2. Previous loading history of the foundation materials, i.e., history of deposition and erosion, groundwater levels, and glacial or other preloading influences on the soil,

142 349

3. Rock jointing pattern and distribution, depth of weathering, zones of alteration or irregular weathering, and zones of structural weakness composed of crushed or disturbed materials such as slickensides, shears, joints, fractures, faults, folds, or a combination of these features. Especially note seams and lenses of weak materials such as clays and weathered shales,

4. Unrelieved residual stresses in bedrock, and

5. Rocks or soils that may be hazardous, or may become hazardous, to the plant because of their lack of consolidation or induration, variability, high water content, solubility, or undesirable response to natural or induced site conditions.

2.5.4.2 Properties of Subsurface Materials. Describe in detail the static and dynamic engineering properties of the materials underlying the site. The classification and engineering properties of soils and rocks should be determined by testing techniques defined by accepted standards such as ASTM and AASHO, or in manuals of practice issued by the Army Corps of Engineers and the Bureau of Reclamation. The determination of dynamic or special engineering properties should be by accepted state-of-the-art methods such as those described in professional geotechnical journals. Reported properties of foundation materials should be supported by field and laboratory test records. Furnish data to justify and demonstrate the selection of design parameters. These data should be sufficient to permit the staff to make an independent interpretation and evaluation of design parameters. Furnish summaries of the physical (static and dynamic), index, and chemical properties of materials. Information provided should include grain-size distribution (graphic representation), consolidation data, mineralogy, natural moisture content, Atterberg limits, unit weights, shear strength, relative density, overconsolidation ratio, ion exchange capacity, sensitivity, swelling, shear modulus, damping, Poisson's ratio, bulk modulus, cyclic strength, and seismic wave velocities.

2.5.4.3 Exploration. Discuss the type, quantity, extent, and purpose of all explorations. Provide plot plans that graphically show the location of all site explorations such as boring, trenches, borrow pits, seismic lines, piezometers, wells, geologic profiles, and the limits of required excavations. The locations of the safety-related facilities should be superimposed on the plot plan. Also, furnish selective geologic sections and profiles that indicate the location of borings and other site exploration features, groundwater elevations, and final foundation grades. The location of safety-related foundations should be superimposed on these sections and profiles.

Logs of all borings and test pits should be provided. Furnish logs and maps of exploratory trenches in the PSAR and geologic maps and photographs of the excavations for the facilities of the nuclear power plant in the FSAR.

142 350

2.5.4.4 Geophysical Surveys. Results of compressional and shear wave velocity surveys performed to evaluate the occurrence and characteristics of the foundation soils and rocks should be provided in tables and profiles. Discuss other geophysical methods used to define foundation conditions.

2.5.4.5 Excavations and Backfill. The following data concerning excavation, backfill, and earthwork at the site should be discussed:

1. The extent (horizontally and vertically) of all Seismic Category I excavations, fills, and slopes. The locations and limits of excavations, fills, and backfills should be shown on plot plans and on geologic sections and profiles.

2. The dewatering and excavation methods to be used. Evaluate how these will affect the quality and condition of foundation materials. Discuss the need and proposed measures for foundation protection and treatment after excavation. Also discuss proposed quality control and quality assurance programs related to foundation excavation, and subsequent protection and treatment. Discuss measures to monitor foundation rebound and heave.

3. The sources and quantities of backfill and borrow. Describe exploration and laboratory studies and the static and dynamic engineering properties of these materials in the same fashion as described in Sections 2.5.4.2 and 2.5.4.3. Provide the plans for field test fills and identify the material and placement specification proposed in the PSAR. Include grain size bands, moisture control, and compaction requirements. Results of test fills should be included in the FSAR.

2.5.4.6 Groundwater Conditions. The analysis of groundwater at the site should include the following points:

1. A discussion of groundwater conditions relative to the stability of the safety-related nuclear power plant facilities,

2. A discussion of design criteria for the control of groundwater levels or collection and control of seepage,

3. Requirements for dewatering during construction and a discussion of how dewatering will be accomplished,

4. Description and interpretation of actual groundwater conditions experienced during construction (FSAR),

5. Records of field and laboratory permeability tests,

6. History of groundwater fluctuations, including those due to flooding, and projected variances in the groundwater levels during the life of the plant,

142 351

7. Information related to the periodic monitoring of local wells and piezometers,

8. Direction of groundwater flow, gradients, and velocities.

9. Discussion of or reference to the groundwater monitoring program during the life of the plant to assess the potential for subsidence.

2.5.4.7 Response of Soil and Rock to Dynamic Loading. Furnish analyses of the responses of the soil and rock to dynamic and seismic loading conditions. Discuss the testing performed and test results. Justify selected design values used for dynamic response analyses. Justify the methods of analyses used and indicate the results of analyses. Identify computer programs used and provide abstracts. Soil-structure interaction analyses should be described in this section or cross-referenced from Section 3.7.2.4. Buried pipelines and earthworks should also be included in this section.

2.5.4.8 Liquefaction Potential. If the foundation materials at the site adjacent to and under safety-related structures are saturated soils or soils that have a potential for becoming saturated, an appropriate state-of-the-art analysis of the potential for liquefaction occurring at the site should be provided. The method of analysis should be determined on the basis of actual site conditions, the properties of the plant facilities, and the earthquake and seismic design requirement.

2.5.4.9 Earthquake Design Basis. A summary should be provided of the derivation of the OBE and SSE, including references to Sections 2.5.2.6 and 2.5.2.7. Justify the selection of earthquakes for liquefaction and seismic response analysis of earthworks.

2.5.4.10 Static Stability. The stability of all safety-related facilities should be analyzed for static loading conditions. Foundation rebound, settlement, differential settlement, and bearing capacity should be analyzed under the design loads of fills and plant facilities. A discussion and evaluation of lateral earth pressures and hydrostatic groundwater loads acting on plant facilities should be included in this section. Field and laboratory test results should be discussed. Design parameters used in stability analyses should be discussed and justified. Sufficient data and analyses should be provided so that the staff may make an independent interpretation and evaluation. Results of stability analyses should be presented in the PSAR and confirmed with as-built data in the FSAR.

2.5.4.11 Design Criteria. Provide a brief discussion of the design criteria and methods of design used in the stability studies of all safety-related facilities. Identify required and computed factors of safety, assumptions, and conservatisms in each analysis. Provide references. Explain and verify computer analyses used.

2.5.4.12 Techniques to Improve Subsurface Conditions. Discuss and provide specifications for measures to improve foundations such as grouting, vibroflotation, dental work, rock bolting, and anchors. A verification program designed to permit a thorough evaluation of the effectiveness of foundation improvement measures should also be discussed.

2.5.4.13 Subsurface Instrumentation. Instrumentation for the surveillance of foundations for safety-related structures should be presented in this section. Indicate the type, location, and purpose of each instrument and provide significant details of installation methods. Provide a schedule for installing and reading all proposed instruments and for the interpretation of the data obtained (PSAR). Results and analyses should be presented in the FSAR.

2.5.4.14 Construction Notes. Significant construction problems should be discussed. Discuss changes in design details or construction procedures that became necessary during construction (FSAR).

2.5.5 Stability of Slopes

Information should be presented concerning the static and dynamic stability of all soil or rock slopes, both natural and man-made, the failure of which could adversely affect the safety of the nuclear power plant. This information should include a thorough evaluation of site conditions, geologic features, the engineering properties of the materials comprising the slope and its foundation. The stability of slopes should be evaluated using classic and contemporary methods of analyses. The evaluation should include, whenever possible, comparative field performance of similar slopes. All information related to defining site conditions, geologic features, the engineering properties of materials, and design criteria should be of the same scope as that provided under Section 2.5.4. Cross-references may be used where appropriate. The stability evaluation of man-made slopes should include summary data and a discussion of construction procedures, record testing, and instrumentation monitoring to ensure high quality earthwork.

2.5.5.1 Slope Characteristics. Describe and illustrate slopes and related site features in detail. Provide a plan showing the limits of cuts, fills, or natural undisturbed slopes and show their relation and orientation relative to plant facilities. Benches, retaining walls, bulkheads, jetties, and slope protection should be clearly identified. Provide detailed cross sections and profiles of all slopes and their foundations. Discuss exploration programs and local geologic features. Describe the groundwater and seepage conditions that exist and those assumed for analysis purposes. The type, quantity, extent, and purpose of exploration should be described and the location of borings, test pits, and trenches should be shown on all drawings.

Discuss sampling methods used. Identify material types and the static and dynamic engineering properties of the soil and rock materials comprising the slopes and their foundations. Identify the presence of any weak zones,

142 353

such as seams or lenses of clay, mylonites, or potentially liquefiable materials. Discuss and present results of the field and laboratory testing programs and justify selected design strengths.

2.5.5.2 Design Criteria and Analyses. The design criteria for the stability and design of all safety-related and Seismic Category I slopes should be described. Valid static and dynamic analyses should be presented to demonstrate the reliable performance of these slopes throughout the lifetime of the plant. Describe the methods used for static and dynamic analyses and indicate reasons for selecting them. Indicate assumptions and design cases analyzed with computed factors of safety. Present the results of stability analyses in tables identifying design cases analyzed, strength assumptions for materials, and type of failure surface. Assumed failure surfaces should be graphically shown on cross sections and appropriately identified on both the tables and sections. Explain and justify computer analyses; provide an abstract of computer programs used.

2.5.5.3 Logs of Borings. Present the logs of borings, test pits and trenches that were completed for the evaluation of slopes, foundations, and borrow materials to be used for slopes. Logs should indicate elevations, depths, soil and rock classification information, groundwater levels, exploration and sampling method, recovery, RQD, and blow counts from standard penetration tests. Discuss drilling and sampling procedures and indicate where samples were taken on the logs.

2.5.5.4 Compacted Fill. In this section, provide information related to material, placement, and compaction specifications for fill (soil and/or rock) required to construct slopes such as canal or channel slopes, breakwaters, and jetties. Planned construction procedures and control of earthworks should be thoroughly described. Information necessary is similar to that outlined in Section 2.5.4.5. Quality control techniques and documentation during and following construction should be discussed and referenced to quality assurance sections of the SAR.

2.5.6 Embankments and Dams

This section should include information related to the investigation, engineering design, proposed construction, and performance of all earth, rock, or earth and rock fill embankments used for plant flood protection or for impounding cooling water required for the operation of the nuclear power plant. The format given below may be used for both Seismic Category and safety-related embankments, the failure of which could threaten the public health and safety. The following information should be included: (1) the purpose and location of the embankment and appurtenant structures (spillways, outlet works, etc.), (2) specific geologic features of the site, (3) engineering properties of the bedrock and foundation and embankment soils, (4) design assumptions, data, analyses, and discussions on foundation treatment and embankment design, (5) any special construction requirements, and (6) proposed instrumentation and performance monitoring systems and programs.

142 354

Embankment design studies should include an evaluation of the performance of the embankment on the basis of instrumentation monitored during construction and during the initial reservoir filling. Information related to the evaluation of embankment performance should be provided in the FSAR.

Any significant event such as an earthquake or flood that occurs during construction or during the initial reservoir filling should be documented in the FSAR together with all information related to the performance of the embankment and observed behavior within its foundation and abutments during the event.

Photographs showing general views of damsite (before, during, and after construction), foundation stripping and treatment (FSAR), construction equipment and activities (FSAR), instrumentation devices and installation work (FSAR), and special items should be provided.

Embankment zone placement quantities, a comparison of embankment zone design placement requirements with a summary of field control test data results (FSAR), and a comparison of embankment shear strength design assumptions with a summary of record control shear strength test results (FSAR) should be tabulated.

The following drawings should be provided:

1. General plan with vicinity map,

2. Large-scale embankment plan with boring and instrumentation locations shown,

3. Geologic profile embankment axis, control structure axis, and spillway axis,

4. Embankment cross sections with instrumentation shown,

5. Embankment details,

6. Embankment foundation excavation plan,

7. Embankment and foundation design shear strength test data graphic summaries with selected design values shown,

8. Embankment slope stability cross sections with design assumptions, critical failure planes, and factors of safety shown,

9. Embankment slope stability reevaluation, if necessary (FSAR),

10. Embankment seepage control design with assumptions, section, and selected design shown,

142 355

11. Relief well profile with the quantities of flow measured at various depths in the relief wells shown (FSAR),

12. Plot of pool elevation versus total relief well discharge quantities (FSAR),

13. Distribution of field control test locations. For each zone tested, plot a profile parallel to the axis with field control test data plotted at the locations sampled.

14. Instrumentation installation details,

15. Interpretations of instrumentation data.

a. Settlement profile or contour plan,

b. Alignment profiles of measured movements,

c. Embankment section with embankment and foundation pore pressure contours. May be necessary to plot contour diagrams at various dates.

d. Embankment sections showing phreatic surface through foundation,

e. Profile in relief well line showing well and piezometer locations and measured and design heads.

2.5.6.1 General. The purpose of the embankment, including natural and severe conditions under which it is to function, should be stated. Identify the reasons for selecting the proposed location within the site. General design features, including planned water control structures, should be discussed.

2.5.6.2 Exploration. Discuss exploration and the local geologic features of the proposed embankment site, and relate these features to the plant site in general. The type, quantity, extent, and purpose of the underground exploration program should be provided. Exploration and sampling methods used should be discussed.

2.5.6.3 Foundation and Abutment Treatment. Discuss the need for, and justify the selection of the types of foundation and abutment treatment such as grouting, cutoff trenches, and dental treatment. Evaluate and report the effectiveness of the completed foundation and abutment treatment programs in the FSAR. The areal extent and depth limits of treatment should be shown on plot plans. Discuss the construction procedures to be employed, and estimate the construction quantities involved.

2.5.6.4 Embankment. Present the general embankment features, including height, slopes, zoning, material properties (including borrow and foundation), sources of materials, and location and usage of materials

142 356

in the embankment. Slope protection design, material properties, and placement methods should be presented. Discuss consolidation testing results, embankment settlement, and overbuild.

Compaction test results on laboratory test specimens and on test fills in the field should be discussed, as well as field control to be specified for the foundation preparation and protection and also for placement of fill, including material requirements, placement conditions, moisture control, and compaction. Also, discuss protection required of fill surfaces and stockpiles during construction, compaction equipment to be used, and any special fill placement activities required. The FSAR should document compliance with specifications related to foundation preparation and also with material specifications and fill placement requirements. Significant or unusual construction activities and problems should also be documented in the FSAR.

2.5.6.5 Slope Stability. For both the foundation and embankment materials, discuss the shear testing performed, shear test data results, selected design strengths, reasons for selecting the method of slope stability analysis used, and the results of design cases analyzed for the embankment constructed.

2.5.6.6 Seepage Control. Exploration and testing performed to determine assumptions used for seepage analyses should be discussed. Present design assumptions, results of design analyses, and reasons for the seepage control design selected. Special construction requirements as well as activities related to the final construction of seepage control features should be discussed in the FSAR.

2.5.6.7 Diversion and Closure. Programs needed for the care and diversion of water during construction should be discussed, including the need for cofferdams, techniques used to dewater excavations, and any expected problems or difficulties. Discuss the proposed diversion and closure construction sequence. Relate actual construction experiences in the FSAR.

2.5.6.8 Performance Monitoring. The overall instrumentation plan and the purpose of each set of instruments should be discussed, as well as the different kinds of instruments, special instruments, and significant details for installation of instruments. Provide the program for periodic monitoring of instrumentation and periodic inspection of the embankment and appurtenant structures.

2.5.6.9 Construction Notes (FSAR). Significant embankment construction history should be provided. Discuss changes in design details or construction procedures that became necessary during construction.

2.5.6.10 Operational Notes. Embankment performance history since completion of construction should be provided in the FSAR.

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

This chapter of the SAR should identify, describe, and discuss the principal architectural and engineering design of those structures, components, equipment, and systems important to safety.

3.1 Conformance with NRC General Design Criteria

This section should briefly discuss the extent to which the design criteria for the plant structures, systems, and components important to safety meet the NRC "General Design Criteria for Nuclear Power Plants" specified in Appendix A to 10 CFR Part 50. For each criterion, a summary should be provided to show how the principal design features meet the criterion. Any exceptions to criteria should be identified and the justification for each exception should be discussed. In the discussion of each criterion, the sections of the SAR where more detailed information is presented to demonstrate compliance with or exceptions to the criterion should be referenced.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

This section should identify those structures, systems, and components important to safety that are designed to withstand the effects of a Safe Shutdown Earthquake (see Section 2.5) and remain functional. 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 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 Part 100.

Guidance for determining the seismic classification of structures, systems, and components is provided in Regulatory Guide 1.29, "Seismic Design Classification." These plant features, including their foundations and supports, designed to remain functional in the event of a Safe Shutdown Earthquake are designated as Seismic Category I. This section should indicate if the recommendations of Regulatory Guide 1.29 are being followed and provide a list of all Seismic Category I items. If only portions of structures and systems are Seismic Category I, they should be listed and, where necessary for clarity, the boundaries of the Seismic Category I portions should be shown on piping and instrumentation diagrams. Where there are differences with Regulatory Guide 1.29, they should be identified and a discussion of the proposed classification should be included.

142 358

All structures, systems, and components or portions thereof, which are intended to be designed for an Operating Basis Earthquake (OBE), should be listed or otherwise clearly identified.

3.2.2 System Quality Group Classifications

This section should identify those fluid systems or portions of fluid systems important to safety and the industry codes and standards applicable to each pressure-retaining component in the systems.

Section 50.55a of 10 CFR Part 50 specifies quality requirements for the reactor coolant pressure boundary, and Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," describes a quality group classification system and relates it to industry codes for water- and steam-containing fluid systems. The section should indicate the extent to which the recommendations of Regulatory Guide 1.26 are followed. Where there are differences, they should be identified and a discussion included justifying each proposed quality group classification in terms of the reliance placed on these systems:

1. To prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary,

2. To permit shutdown of the reactor and maintenance in the safe shutdown condition, and

3. To contain radioactive material.

In such cases, the proposed design features and measures that would be applied to attain a quality level equivalent to the level of the above classifications should be specified, including the quality assurance programs that would be implemented. The section should contain group classification boundaries of each safety-related system. The classifications should be noted at valves or other appropriate locations in each fluid system where the respective classification changes in terms of the NRC group classification letters, for example, from A to B, B to C, C to D as well as other combinations, or alternately, in terms of corresponding classification notations that can be referenced with those classification groups in Regulatory Guide 1.26.

3.3 Wind and Tornado Loadings

3.3.1 Wind Loadings

This section should discuss the design wind load on Seismic Category I structures and, in particular, should include the information identified below.

142 359

3.3.1.1 Design Wind Velocity. The design wind velocity and its recurrence interval, the vertical velocity profiles, and the applicable gust factors, as described in Section 2.3, should be presented here for information.

3.3.1.2 Determination of Applied Forces. The procedures used to transform the wind velocity into an effective pressure applied to exposed surfaces of structures should be described. Wind force distribution and shape coefficients being applied should be included.

3.3.2 Tornado Loadings

This section should discuss the design basis tornado loadings on structures that must be designed for tornadoes. It should include the information identified below.

3.3.2.1 Applicable Design Parameters. The design parameters applicable to the design basis tornado should be presented here for information. The translational velocity, the tangential velocity, the pressure differential and its associated time interval, and the spectrum and pertinent characteristics of tornado-generated missiles should be included. Material covered in Sections 2.3 and 3.5.1 may be incorporated by reference.

3.3.2.2 Determination of Forces on Structures. The procedures used to transform the tornado loadings into effective loads on structures should be described. The following information should be included:

1. The procedures used for transforming the tornado wind into an effective pressure on exposed surfaces of structures. Shape coefficients and pressure distribution on flat surfaces and round structures such as containments should also be included.

2. If venting of a structure is used, the procedures employed for transforming the tornado-generated differential pressure into an effective reduced pressure.

3. The procedures used for transforming the tornado-generated missile loadings, which are considered impactive dynamic loads, into effective loads. Material included in Section 3.5.3 may be referenced in this section.

4. The various combinations of the above individual loadings that will produce the most adverse total tornado effect on structures.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads. This section should present information to show that the failure of any structure or component not designed for tornado loads will not affect the ability of other structures to perform their intended safety functions.

142 360

3.4 Water Level (Flood) Design

This section should discuss the flood and/or the highest ground water level design for Seismic Category I structures and components including the following information.

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Category I Structures. The flood protection measures for Seismic Category I structures, systems, and components should be described and include the following:

1. Identify the safety-related systems and components that should be protected against floods (see Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," and Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants"), and show the relationship to design flood levels and conditions defined in Section 2.4 (include station drawings).*

2. Describe the structures that house safety-related equipment, including an identification of exterior or access openings and penetrations that are below the design flood levels.*

3. If flood protection is required, discuss the means of providing flood protection (e.g., pumping systems, stoplogs, watertight doors, and drainage systems) for equipment that may be vulnerable because of its location and the protection provided to cope with potential inleakage from such phenomena as cracks in structure walls, leaking water stops, and effects of wind wave action (including spray). Identify on plant layout drawings individual compartments or cubicles that house safety-related equipment and that act as positive barriers against possible flooding.

4. Describe the procedures required (see regulatory position 2 of Regulatory Guide 1.59 and regulatory position 2 of Regulatory Guide 1.102) and implementation times available to bring the reactor to a cold shutdown for the flood conditions identified in Section 2.4.14. These procedures and times should be compared with the procedures and times required to implement flood protection requirements identified in Section 2.4.14.

5. Identify those safety-related systems or components, if any, that are capable of normal function while completely or partially flooded.

3.4.1.2 Permanent Dewatering System. This section should describe any permanent dewatering system provided to protect safety-related structures, systems, or components from the effects of ground water. The following information should be included:

* The details discussed herein should be consistent with Sections 2.4.1.1, 2.4.2.2, and 2.4.10.

142 361

1. A summary description of the dewatering system. All major subsystems such as the active discharge subsystem and the passive collection and drainage subsystem should be described.

2. The design bases for the functional performance requirements for each subsystem, along with the bases for selecting the system operating parameters.

3. A safety evauation demonstrating how the system satisfies the design bases, the system's capability to withstand design basis events, and its capability to perform its safety function assuming a single active failure with the loss of offsite power. Protection against single failure should be evaluated in terms of piping arrangement and layout, selection of valve types and location, redundancy of various system components, redundancy of power supplies, redundant sources of actuation signals, and redundancy of instrumentation. The safety evaluation should also demonstrate that the dewatering system is protected from the effects of pipe breaks and missiles.

4. The testing and inspection to be performed to verify that the system capability and reliability are met and the instrumentation and control necessary for proper operation of the system.

5. The information provided in this section of the SAR should be consistent with the information provided in Sections 2.4 and 2.5, which should be referenced as appropriate.

3.4.2 Analytical and Test Procedures

Describe the methods and procedures by which the static and dynamic effects of the design basis flood conditions or design basis ground water conditions identified in Section 2.4 are applied to safety-related structures, systems, and components. Summarize for each safety-related structure, system, and component that may be so affected, the design basis static and dynamic loadings, including consideration of hydrostatic loadings, equivalent hydrostatic dynamically induced loadings, coincident wind loadings, and the static and dynamic effects on foundation properties (Section 2.5).

If physical models are used to predict prototype performance of hydraulic structures and systems, information on such model testing should be included herein (see Regulatory Guide 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants").

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment). The design bases for the structures, systems (or portions of systems), and

143 001

components that are to be protected against damage from internally generated missiles outside containment should be provided. Missiles associated with overspeed failures of rotating components and with failures of high-pressure system components should be considered. The design bases should consider the design features provided for either continued safe operation or shutdown during all operating conditions, operational transients, and postulated accident conditions.

A tabulation showing the safety-related structures, systems, and components outside containment required for safe shutdown of the reactor under all conditions of plant operation should be provided and, as a minimum, should include the following:

1. Locations of the structures, systems, or components.

2. Applicable seismic category and quality group classifications (may be referenced from Section 3.2).

3. Sections in the SAR where descriptions of the items may be found.

4. Reference drawings or piping and instrumentation diagrams where applicable (may be referenced from other sections of the SAR).

5. Identification of missiles to be protected against, their source, and the bases for selection.

6. Missile protection provided.

The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles should be evaluated.

3.5.1.2 Internally Generated Missiles (Inside Containment). All plant structures, systems, and components inside containment whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown to a cold condition assuming an additional single failure should be identified. The separation and independence of those structures, systems, and components protected by redundancy rather than physical barriers against very low probability missile strikes should be clearly demonstrated. The structures, systems, and components protected by physical barriers should be identified. Missiles associated with overspeed failures of rotating components, with primary and secondary failures of high-pressure system components, and those due to gravitational effects should be identified.

A tabulation showing the safety-related structures, systems, and components inside containment required for safe shutdown of the reactor under all conditions of the plant operation, including operational transients and postulated accident conditions, should be provided and, as a minimum, should include the following:

143 002

1. Location of the structure, system, or component.

2. Identification of missiles to be protected against, their source, and the bases for selection.

3. Missile protection provided.

The ability of the structures, systems, and components to withstand the effects of selected internally generated missiles should be evaluated.

3.5.1.3 Turbine Missiles. Information should be provided on the following topics:

1. Turbine Placement and Orientation. Plant layout drawings should indicate clearly turbine placement and orientation. Plan and elevation views should have appropriate indication of the ± 25-degree low-trajectory turbine missile ejection zone with respect to the low-pressure turbine wheels for each turbine unit "within reach" of the plant structures. Target areas should be indicated clearly on plan and elevation views with respect to all systems identified in Section 3.5.2.

2. Missile Identification and Characteristics. Description of postulated turbine missiles should include missile properties such as mass, shape, cross-sectional areas, ranged turbine exit speeds, and range of turbine exit angles. Mathematical models used in the analysis of such items as missile selection, turbine casing penetration, and missile trajectories should be included. A description of the analytical models used to determine the characteristics of the selected missiles, including any assumptions, should be included.

3. Target Description. Structures and equipment identified in Section 3.5.2, if within the low-trajectory turbine missile strike zones described in item 1, should be identified in dimensioned plan and elevation drawings. Safety-related equipment occupying small portions of a room or a structure should be indicated individually (e.g., batteries, switchgear cabinets, isolation valves). Separation distances and/or separation barriers should be indicated with respect to redundant equipment.

4. Probability Analysis. An analysis of strike probabilities for low-trajectory turbine missiles with respect to plant systems identified in Section 3.5.2 should be provided. If the analytical methods are described by referencing other documents, a brief summary outline of the method, including sample calculations, should be provided. All assumptions used in the analysis should be identified, and the bases supporting them should be discussed.

Numerical results of the analysis should be presented in tabular form, listing the individual strike probabilities for each vital area with respect to design and destructive overspeed turbine missiles. The data should be resolved into strike probability contributions from each turbine unit (including nonnuclear units) on or in the vicinity of the site.

In the case of destructive overspeed, an analysis should be presented justifying the assumption of only one disc failure. Turbine overspeed acceleration characteristics, statistical distribution of destructive overspeed failure speeds, and related information should be considered in the evaluation of the probability of second wheel failure during the interval of physical disassembly caused by the first failure.

5. Turbine Overspeed Protection. A description of the turbine overspeed protection system in terms of redundancy, diversity, component reliability, and testing procedures should be provided.

6. Turbine Valve Testing. A discussion of the turbine valve testing environment should cover such items as test frequency, power level, pressure difference across the steam valve(s), and other parameters pertinent to overspeed protection.

7. Turbine Characteristics. Turbine data pertinent to the evaluation of its failure characteristics should include a description of its overall configuration, major components (e.g., steam valves, reheaters, etc.), rotor materials and their properties, steam environment (e.g., pressure, temperature, quality, chemistry), and other appropriate properties. Turbine operational and transient characteristics should be described, including turbine startup and trip environments, as well as its overspeed parameters (e.g., time to 180% overspeed from loss of 100% power load).

3.5.1.4 Missiles Generated by Natural Phenomena. Identify all missiles generated as a result of natural phenomena (e.g., tornadoes and floods) in the vicinity of the plant. For selected missiles, specify the origin, dimensions, mass, energy, velocity, and any other parameters required to determine missile penetration.

The structures and/or barriers used for missile protection should be tabulated. The table should contain the following information:

1. Systems or components that are protected by the structure/barrier.

2. Concrete thickness and strength for walls, roofs, and floors used for missile protection and the curing time on which the strength is based.

3.5.1.5 Missiles Generated by Events Near the Site. Identify all missile sources resulting from accidental explosions in the vicinity of the site. The presence of and operations at nearby industrial, transportation, and military facilities should be considered. The following missile sources should be considered with respect to the site:

1. Train explosions (including rocket effects),

2. Truck explosions,

3. Ship or barge explosions,

4. Industrial facilities,

5. Pipeline explosions, and

6. Military facilities.

Missiles from each type of source should be characterized in terms of dimensions, mass, energy, velocity, trajectory, and energy density. (Aircraft crashes should be analyzed in Section 3.5.1.5.)

3.5.1.6 Aircraft Hazards. An aircraft hazard analysis should be provided for each of the following:

1. Federal airways or airport approaches passing within 2 miles of the nuclear facility.

2. All airports located within 5 miles of the site.

3. Airports with projected operations greater than $500d^2$ movements per year located within 10 miles of the site and greater than $1000d^2$ outside 10 miles, where d is the distance in miles from the site.

4. Military installations or any airspace usage that might present a hazard to the site. For some uses such as practice bombing ranges, it may be necessary to evaluate uses as far as 20 miles from the site.

The analyses should provide an estimate of the probability of an aircraft accident with consequences worse than those of the design basis accident. Hazards to the plant may be divided into accidents resulting in structural damage and accidents involving fire. These analyses should be based on the projected traffic for the facilities, the aircraft accident statistics provided in Section 2.2, and the critical areas described in Section 3.5.2.

All the parameters used in these analyses should be explicitly justified. Wherever a range of values is obtained for a given parameter, it should be plainly indicated and the most conservative value used. Justification for all assumptions made should also be clearly stated.

Conclusions on the aircraft, if any, that are to be selected as design basis impact events should be stated and the rationale for the choice clearly set forth. The whole aircraft or parts thereof should be characterized in terms of dimensions, mass (including variations along the length of the aircraft), energy, velocity, trajectory, and energy

143 005

density. Resultant loading curves on structures should be presented in Section 3.5.3.

3.5.2 Structures, Systems, and Components To Be Protected from Externally Generated Missiles

All plant structures, systems, and components whose failure could lead to offsite radiological consequences or that are required to shut down the reactor and maintain it in a safe condition assuming an additional single failure should be identified. It should be demonstrated that such safety-related structures, systems, and components are adequately protected against very low probability missile strikes by physical barriers or protective structures. Missiles that should be considered are identified in Section 3.5.1. Protective structures and barriers should be identified on plant arrangement and elevation drawings and in the system and component classification tables.

3.5.3 Barrier Design Procedures

The procedures by which each structure or barrier will be designed to resist the missile hazards previously described should be presented; the following should be included:

1. Procedures utilized (a) to predict local damage in the impact area, including estimation of the depth of penetration, (b) to estimate barrier thickness required to prevent perforation, and (c) in the case of concrete barriers to predict the potential for generating secondary missiles by spalling and scabbing effects, and

2. Procedures utilized for the prediction of the overall response of the barrier and portions thereof to missile impact. This includes assumptions on acceptable ductility ratios and estimates of forces, moments, and shears induced in the barrier by the impact force of the missile.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

This section should describe design bases (PSAR) and design measures (FSAR) to ensure that the containment vessel and all essential equipment inside or outside the containment, including components of the reactor coolant pressure boundary, have been adequately protected against the effects of blowdown jet and reactive forces and pipe whip resulting from postulated rupture of piping located either inside or outside of containment. The following specific information should be included.

3.6.1 Postulated Piping Failures in Fluid Systems Outside of Containment

3.6.1.1 Design Bases (PSAR). Systems or components important to plant safety or shutdown that are located proximate to high- or moderate-energy piping systems and that are susceptible to the consequences of failures of these piping systems should be identified. The identification

143 006

should be related to predetermined piping failure locations in accordance with Section 3.6.2. Typical piping runs with failure points indicated on drawings should be provided. The identification of affected components should also include limiting acceptable conditions, i.e., those conditions for which operation of the component will not be precluded. The design approach taken to protect the systems and components identified above should be indicated.

3.6.1.2 Description (FSAR). Provide a listing of high- and moderate-energy lines. In the case where physical arrangement of the piping system provides the required protection, a description of the layout of all systems should be submitted. In the case where the high- or moderate-energy piping systems have been enclosed in structures or compartments to protect nearby essential systems or components, descriptions and pressure rise analyses should be provided to verify the structural adequacy of such enclosures. An analysis of the potential effects of secondary missiles on the components should also be provided. If failure of or leakage from high- or moderate-energy lines affect nearby safety features or results in the transport of a steam environment to other rooms or compartments in the facility, an analysis should be provided of the effects of the environment on the operation of the affected equipment or systems. In the case of the control room, analyses should be provided to verify that habitability will be ensured.

3.6.1.3 Safety Evaluation (FSAR). The results of failure mode and effects analyses should be provided to verify that the consequences of failures of high- and moderate-energy lines do not affect the ability to safely shut the plant down. The analyses should include consideration of single active component failures occurring in required systems concurrently with the postulated event.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

This section should describe the design bases for locating postulated breaks and cracks in piping inside and outside of containment, the procedures used to define the jet thrust reaction at the break or crack location, and the jet impingement loading on adjacent safety-related structures, equipment, systems, and components.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration (PSAR). The criteria should be provided for the location and configuration of postulated breaks and cracks in those high- and moderate-energy piping systems for which separation or enclosure cannot be achieved. In the case of containment penetration piping, in addition to the material requested above, the details of the containment penetration identifying all process pipe welds, access for inservice inspection of welds, points of fixity, and points of geometric discontinuity should be provided.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models (PSAR). The methods used to define the forcing functions to be used for the pipe whip dynamic analyses should be described. The description should include direction, thrust coefficients, rise time, magnitude, duration, and initial conditions that adequately represent the jet stream dynamics and the system pressure differences. Pipe restraint rebound effects should be included if appropriate.

Diagrams of typical mathematical models used for the dynamic response analysis should be provided. All dynamic amplification factors to be used should be presented and justified.

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability (PSAR). The method of analysis that will be used to evaluate the jet impingement effects and loading effects applicable to components and systems resulting from postulated pipe breaks and cracks should be provided. In addition, provide the analytical methods used to verify the integrity of mechanical components under pipe rupture loads. In the case of piping systems where pipe whip restraints are included, the loading combinations and the design criteria for the restraints should be provided along with a descript typical of the restraint configuration to be used.

3.6.2.4 Guard Pipe Assembly Design Criteria (PSAR). The details of protective assemblies or guard pipes (a guard pipe is a device to limit pressurization of the space between dual barriers of certain containments to acceptable levels) to be used for piping penetrations of containment areas should be provided. Discuss whether such protective assemblies serve to provide an extension of containment, prevent overpressurization, or both.

The use of moment-limiting restraints at the extremities or within the protective assembly should be indicated. The following should be provided:

1. The criteria for the design of the process pipe within the protective assembly. Include type of material (seamless or welded), allowable stress level, and loading combinations.

2. The design criteria to be used for flued heads and bellows expansion joints.

3. The design criteria applicable to the guard pipe that is used with the assembly.

4. A description of the method of providing access and the location of such access openings to permit periodic examinations of all process pipe welds within the protective assembly as required by the plant inservice inspection program (refer to Section 5.2.4 for ASME Class 1 systems and Section 6.6 for ASME Class 2 and 3 systems).

143 008

3.6.2.5 Material To Be Submitted for the Operating License Review (FSAR). A summary of the dynamic analyses applicable to high- and moderate-energy piping systems and associated supports that determine the loadings resulting from postulated pipe breaks and cracks should be presented. The following should be included:

1. The implementation of criteria for defining pipe break and crack locations and configurations. Provide the locations and number of design basis breaks and cracks on which the dynamic analyses are based. Also provide the postulated rupture orientation, such as the circumferential and/or longitudinal break(s), for each postulated design basis break location.

2. The implementation of criteria dealing with special features such as augmented inservice inspection program or the use of special protective devices such as pipe whip restraints, including diagrams showing their final configurations, locations, and orientations in relation to break locations in each piping system.

3. The acceptability of the analysis results, including the jet thrust and impingement functions and the pipe whip dynamic effects.

4. The design adequacy of systems, components, and component supports to ensure that their design-intended functions will not be impaired to an unacceptable level of integrity or operability as a result of pipe whip loading or jet impingement loading.

5. The implementation of the criteria relating to protective assembly design, including the final design, location of restraints, stress levels for various plant operating conditions for the process pipe, flued heads, bellows expansion joints, and guard pipes. Present the final design and arrangement of the access openings that are used to examine all process pipe welds within such protective assemblies to meet the requirements of the plant inservice inspection program.

3.7 Seismic Design

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra. Design response spectra (Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE)) should be provided to permit comparison with Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," which provides acceptable design response spectra. The basis for any response spectra that differ from the spectra given in Regulatory Guide 1.60 should be included. The response spectra applied at the finished grade in the free field or at the various foundation locations of Seismic Category I structures should be provided.

3.7.1.2 Design Time History. For the time history analyses, the response spectra derived from the actual or synthetic earthquake time-motion records should be provided. A comparison of the response spectra obtained in the free field at the finished grade level and the foundation level (obtained from an appropriate time history at the base of the soil/structure interaction system) with the design response spectra should be submitted for each of the damping values to be used in the design of structures, systems, and components. Alternatively, if the design response spectra for the OBE and SSE are applied at the foundation levels of Seismic Category I structures in the free field, a comparison of the free-field response spectra at the foundation level (derived from an actual or synthetic time history) with the design response spectra should be provided for each of the damping values to be used in the design. The period intervals at which the spectra values were calculated should be identified.

3.7.1.3 Critical Damping Values. The specific percentage of critical damping values used for Seismic Category I structures, systems, and components and soil should be provided for both the OBE and SSE (e.g., damping values for the type of construction or fabrication such as prestressed concrete and welded pipe) to permit comparison with Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," which provides acceptable damping values. The basis for any proposed damping values that differ from those given in Regulatory Guide 1.61 should be included.

3.7.1.4 Supporting Media for Seismic Category I Structures. A description of the supporting media for each Seismic Category I structure should be provided. Include in this description foundation embedment depth, depth of soil over bedrock, soil layering characteristics, width of the structural foundation, total structural height, and soil properties such as shear wave velocity, shear modulus, and density. This information is needed to permit evaluation of the suitability of using either a finite element or lumped spring approach for soil/structure interaction analysis.

3.7.2 Seismic System Analysis

This section should discuss the seismic system analyses applicable to Seismic Category I structures, systems, and components. The specific information identified in the following sections should be included.

3.7.2.1 Seismic Analysis Methods. The applicable methods of seismic analysis (e.g., modal analysis response spectra, modal analysis time history, equivalent static load) should be identified and described. Descriptions (sketches) of typical mathematical models used to determine the response should be provided. Indicate how the dynamic system analysis method includes in the model consideration of foundation torsion, rocking, and translation. The method chosen for selection of significant modes and adequate number of masses or degrees of freedom should be specified. The manner in which consideration is given in the seismic dynamic analysis to maximum relative displacement among supports should be indicated. In addition, other significant effects that are accounted for in the dynamic

143 010

seismic analysis (e.g., hydrodynamic effects and nonlinear response) should be indicated. If tests or empirical methods are used in lieu of analysis, the testing procedure, load levels, and acceptance bases should also be provided.

3.7.2.2 Natural Frequencies and Response Loads (FSAR). For the operating license review, significant natural frequencies and response loads determined by seismic system analyses should be provided for major Seismic Category I structures. In addition, the response spectra at critical major Seismic Category I elevations and points of support should be specified.

3.7.2.3 Procedure Used for Modeling. The criteria and procedures used for modeling in the seismic system analyses should be provided. Include the criteria and bases used to determine whether a component or structure should be analyzed as part of a system analysis or independently as a subsystem.

3.7.2.4 Soil/Structure Interaction. As applicable, the methods of soil/structure interaction analysis used in the seismic system analysis and their bases should be provided. The following information should be included: (1) the extent of embedment, (2) the depth of soil over rock, and (3) the layering of the soil stratum. If the finite element approach is used, the criteria for determining the location of the bottom boundary and side boundary should be specified. The procedure by which strain-dependent soil properties (e.g., damping and shear modulus) are incorporated in the analysis should also be specified. The material given in Section 3.7.1.4 may be referenced in this section.

If lumped spring methods are used, the parameters used in the analysis should be discussed. Describe the procedures by which strain-dependent soil properties, layering, and variation of soil properties are incorporated into the analysis. The suitability of a lumped spring method used for the particular site conditions should also be discussed.

Any other methods used for soil/structure interaction analysis or the basis for not using soil/structure interaction analysis should be provided.

The procedures used to consider effects of adjacent structures on structural response in soil/structure interaction analysis should be provided.

3.7.2.5 Development of Floor Response Spectra. The procedures for developing floor response spectra considering the three components of earthquake motion should be described. If a modal response spectrum method of analysis is used to develop floor response spectra, the basis for its conservatism and equivalence to a time history method should be provided.

3.7.2.6 Three Components of Earthquake Motion. Indicate the extent to which the procedures for considering the three components of earthquake

143 011

motion in determining the seismic response of structures, systems, and components follow the recommendations of Regulatory Gu de 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

3.7.2.7 Combination of Modal Responses. When a response spectra method is used, a descript on of the procedure for combining modal responses (shears, moments, stresses, deflections, and acceleratio) should be provided. Indicate the extent to which the recommendations of Regulatory Guide 1.9 re followed.

3., 2 3 Interaction of Non-Category I Structures with Seismic Category I Structures. Provide the design criteria used to account for the seismic motion of non-Category I structures or portions thereof in the seismic design of Seismic Category I structures or portions thereof. In addition, describe the design criteria that will be applied to ensure protection of Seismic Category I structures from the structural failure of non-Category I structures due to seismic effects.

3.7.2.9 Effects of Parameter Variations on Floor Response Spectra. The procedures that will be used to consider the effects of expected variations of structural properties, dampings, soil properties, and soil/structure interaction on floor response spectra (e.g., peak width and period coordinates) and time histories should be described.

3.7.2.10 Use of Constant Vertical Static Factors. Where applicable, identify and justify the application of constant static factors as vertical response loads for the seismic design of Seismic Category I structures, systems, and components in lieu of a vertical seismic-system dynamic analysis method.

3.7.2.11 Method Used to Account for Torsional Effects. The method used to consider the torsional effects in the seismic analysis of the Seismic Category I structures should be described. Where applicable, discuss and justify the use of static factors or any other approximate method in lieu of a combined vertical, horizontal, and torsional system dynamic analysis to account for torsional accelerations in the seismic design of Seismic Category I structures.

3.7.2.12 Comparison of Responses (FSAR). For the operating license review where both modal response and time history methods are applied, the responses obtained from both methods at selected points in major Seismic Category I structures should be provided, together with a comparative discussion of the responses.

3.7.2.13 Methods for Seismic Analysis of Dams. A comprehensive description of the analytical methods and procedures that will be used for the seismic system analysis of Seismic Category I dams should be provided. The assumptions made, the boundary conditions used, and the procedures by which strain-dependent soil properties are incorporated in the analysis should be provided.

143 012

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments. A description of the dynamic methods and procedures used to determine Seismic Category I structure overturning moments should be provided.

3.7.2.15 Analysis Procedure for Damping. The analysis procedure used to account for the damping in different elements of the model of a coupled system should be described.

3.7.3 Seismic Subsystem Analysis

This section should discuss the seismic subsystem analyses applicable to Seismic Category I structures, subsystems, and components. The specific information identified in the following sections should be included.

3.7.3.1 Seismic Analysis Methods. Information should be provided as requested in Section 3.7.2.1, but as applied to the Seismic Category I subsystems.

3.7.3.2 Determination of Number of Earthquake Cycles. Describe criteria or procedures that are used to determine the number of earthquake cycles during one seismic event. The maximum number of cycles for which applicable Seismic Category I structures, subsystems, and components are designed should be specified.

3.7.3.3 Procedure Used for Modeling. The criteria and procedures used for modeling for the seismic subsystem analysis should be provided.

3.7.3.4 Basis for Selection of Frequencies. Where applicable, discuss the procedures and criteria used to separate the fundamental frequencies of components and equipment from the forcing frequencies of the support structures.

3.7.3.5 Use of Equivalent Static Load Method of Analysis. The basis for the use of the equivalent static load method of analysis and the procedures used for determining the equivalent static loads should be provided.

3.7.3.6 Three Components of Earthquake Motion. Information should be provided as requested in Section 3.7.2.6, but as applied to the Seismic Category I subsystems.

3.7.3.7 Combination of Modal Responses. Information should be provided as requested in Section 3.7.2.7, but as applied to the Seismic Category I subsystems.

3.7.3.8 Analytical Procedures for Piping. The analytical procedures applicable to seismic analysis piping should be described. Include the methods used to consider differential piping support movements at different support points located within a structure and between structures.

143 013

3.7.3.9 Multiply Supported Equipment Components with Distinct Inputs. The criteria and procedures for seismic analysis of equipment and components supported at different elevations within a building and between buildings with distinct inputs should be described.

3.7.3.10 Use of Constant Vertical Static Factors. Information should be provided as requested in Section 3.7.2.10, but as applied to the Seismic Category I subsystems.

3.7.3.11 Torsional Effects of Eccentric Masses. The criteria and procedures that will be employed to account for the torsional effects of valves and other eccentric masses (e.g., valve operators) in the seismic subsystem analyses should be provided.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels. For buried Seismic Category I piping and tunnels, describe the seismic criteria and methods for considering the compliance of soil media, the settlement due to the earthquake, and the differential movement at support points, penetrations, and entry points into other structures provided with anchors.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping. The analysis procedures used to account for the seismic motion of non-Category I piping systems in the seismic design of Seismic Category I piping should be described.

3.7.3.14 Seismic Analyses for Reactor Internals. The seismic subsystem analyses that will be used in establishing seismic design adequacy of the reactor internals, including fuel elements, control rod assemblies, and control rod drive mechanisms, should be described. The following information should be included:

1. Typical diagrams of dynamic mathematical modeling of the reactor internal structures to be used in the analysis,

2. Damping values and their justification,

3. A description of the methods and procedures that will be used to compute seismic responses,

4. A summary of the results of the dynamic seismic analysis for the operating license review.

3.7.3.15 Analysis Procedure for Damping. Information should be provided as requested in Section 3.7.2.15, but as applied to the Seismic Category I subsystems.

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with Regulatory Guide 1.12. The proposed seismic instrumentation should be discussed and compared with the seismic instrumentation program recommended in Regulatory Guide 1.12, "Instrumentation for Earthquakes." The bases for elements of the proposed program that differ from Regulatory Guide 1.12 should be included.

3.7.4.2 Location and Description of Instrumentation. Seismic instrumentation such as triaxial peak accelerographs, triaxial time history accelerographs, and triaxial response spectrum recorders that will be installed in selected Seismic Category I structures and on selected Seismic Category I components should be described. The bases for selection of these structures and components and the location of instrumentation, as well as the extent to which this instrumentation will be employed to verify the seismic analyses following a seismic event, should be specified.

3.7.4.3 Control Room Operator Notification. The provisions that will be used to inform the control room operator of the value of the peak acceleration level and the input response spectra values shortly after occurrence of an earthquake should be described. The bases for establishing predetermined values for activating the readout of the seismic instrument to the control room operator should be included.

3.7.4.4 Comparison of Measured and Predicted Responses. Provide the criteria and procedures that will be used to compare measured responses of Seismic Category I structures and selected components in the event of an earthquake with the results of the seismic system and subsystem analyses.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

This section should provide the following information on concrete containments and on concrete portions of steel/concrete containments:

1. The physical description.

2. The applicable design codes, standards, and specifications.

3. The loading criteria, including loads and load combinations.

4. The design and analysis procedures.

5. The structural acceptance criteria.

6. The materials, quality control programs, and special construction techniques.

143 015

7. The testing and inservice inspection programs.

3.8.1.1 Description of the Containment. A physical description of the concrete containment or concrete portions of steel/concrete containments should be provided and supplemented with plan and section views sufficient to define the primary structural aspects and elements relied upon to perform the containment function. The geometry of the concrete containment or concrete portions of steel/concrete containments, including plan views at various elevations and sections in at least two orthogonal directions should be provided. The arrangement of the containment and the relationship and interaction of the shell with its surrounding structures and with its interior compartments and floors should be provided to establish the effect that these structures could have upon the design boundary conditions and expected structural behavior of the containment when subjected to design loads.

General descriptive information should be provided for the following:

1. The base foundation slab, including the main reinforcement, the floor liner plate and its anchorage and stiffening system, and the methods by which the interior structures are anchored through the liner plate and into the slab, if applicable.

2. The cylindrical wall, including the main reinforcement and prestressing tendons, if any; the wall liner plate and its anchorage and stiffening system; the major penetrations and the reinforcement surrounding them, including the equipment and personnel hatches and major pipe penetrations; major structural attachments to the wall which penetrate the liner plate, such as beam seats, pipe restraints, and crane brackets; and external supports, if any, attached to the wall to support external structures such as enclosure buildings.

3. The dome and the ring girder, if any, including the main reinforcement and prestressing tendons; the liner plate and its anchorage and stiffening system; and any major attachments to the liner plate made from the inside.

4. Steel components of concrete containments that resist pressure and are not backed by structural concrete should be discussed in Section 3.8.2.

3.8.1.2 Applicable Codes, Standards, and Specifications. Information pertaining to design codes, standards, specifications, regulations, general design criteria, regulatory guides, and other industry standards that are used in the design, fabrication, construction, testing, and inservice inspection of the containment should be provided. The specific edition, date, or addenda of each document should be identified.

3.8.1.3 Loads and Load Combinations. The loads and load combinations that are utilized in the design of the containment should be discussed, with emphasis on the extent of compliance with Article CC-3000 of the

ASME Boiler and Pressure Vessel Code, Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," particularly with respect to the following:

1. Those loads encountered during preoperational testing.

2. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, hydrostatic loads such as those present in pressure-suppression containments utilizing water, and localized transient pressure loads induced by actuation of safety relief valves in BWRs.

3. Those loads that would be sustained in the event of severe environmental conditions, including those that would be induced by the design wind and the Operating Basis Earthquake.

4. Those loads that would be sustained during extreme environmental conditions, including those that would be induced by the Design Basis Tornado and the Safe Shutdown Earthquake.

5. Those loads that would be sustained during abnormal plant conditions, including the design basis loss-of-coolant accident (LOCA). Loads generated by other postulated accidents involving various high-energy pipe ruptures should also be discussed. Loads on the containment induced by such accidents should include associated temperature effects and pressure and localized loads such as jet impingement and associated missile impact. Also, external pressure loads generated by events inside or outside the containment should be discussed.

6. If applicable, those loads that would be encountered after abnormal plant conditions, including flooding of the containment subsequent to a loss-of-coolant accident for the purpose of fuel recovery.

The various combinations of the above loads that should be discussed include testing loads, normal operating loads, normal operating loads with severe environmental loads, normal operating loads with extreme environmental loads, normal operating loads with abnormal loads, normal operating loads with severe environmental and abnormal loads, normal operating loads with extreme environmental and abnormal loads, and post-LOCA flooding loads with severe environmental loads, if applicable.

The loads and load combinations described above are generally applicable to most containments. Other site-related or plant-related design loads may also be applicable. Such loads include those induced by floods, potential aircraft crashes, explosive hazards in proximity to the site, and missiles generated from activities of nearby military installations or from plant-related accidents such as turbine failures. As appropriate, these loads and load combinations should be discussed.

3.8.1.4 Design and Analysis Procedures. The design and analysis procedures utilized for the containment should be described, with emphasis on the extent of compliance with Article CC-3000 of the ASME Code, Section III, Division 2. The assumptions made on the boundary conditions should be described. The treatment of loads, including those that may be nonaxisymmetric, localized, or transient, should be provided. The manner in which creep, shrinkage, and cracking of the concrete are addressed in the analysis and design should be described. Computer programs utilized should be referenced to permit identification with available published programs. Proprietary computer programs should be described in sufficient detail to establish the applicability of the programs and the measures taken to validate the programs with solutions derived from other acceptable programs or with solutions of classical problems. The treatment of the effects of tangential (membrane) shears should be discussed. Information on the evaluation of the effects of expected variation in assumptions and material properties on the analysis results should be provided. The method of analyzing large thickened penetration regions and their effect on the containment behavior should be described. The analysis and design procedures for the liner plate and its anchorage system should be described.

3.8.1.5 Structural Acceptance Criteria. The acceptance criteria relating stresses, strains, gross deformations, and other parameters that identify quantitatively the margins of safety should be specified, with emphasis on the extent of compliance with Article CC-3000 of the ASME Code, Section III, Division 2. The information provided should address the containment as an entire structure, and it should also address the margins of safety related to the major important local areas of the containment, including openings, hatch penetrations, anchorage zones, and other areas important to the safety function. The criteria addressing the various loading combinations should be presented in terms of allowable limits for at least the following major parameters:

1. Compressive stresses in concrete, including membrane, membrane plus bending, and localized stresses.

2. Shear stresses in concrete.

3. Tensile stresses in reinforcement.

4. Tensile stresses in prestressing tendons.

5. Tensile or compressive stress/strain limits in the liner plate, including membrane and membrane plus bending.

6. Force/displacement limits in the liner plate anchors, including those induced by strains in the adjacent concrete.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques. The materials that are used in the construction of the containment should be identified, with emphasis on the extent of compliance with Article CC-2000 of the ASME Code, Section III, Division 2. A summary of the

143 018

engineering properties of the materials should be presented. Among the major materials of construction that should be indicated are the following:

1. The concrete ingredients,

2. The reinforcing bars and splices,

3. The prestressing system,

4. The liner plate,

5. The liner plate anchors and associated hardware,

6. The structural steel used for embedments, such as beam seats and crane brackets, and

7. The corrosion-retarding compounds used for the prestressing tendons.

The quality control program that is proposed for the fabrication and construction of the containment should be described with emphasis on the extent of compliance with Articles CC-4000 and CC-5000 of the ASME Code, Section III, Division 2. The description should show the extent to which the quality control program covers the examination of materials, including tests to determine the physical properties of concrete, reinforcing steel, mechanical splices, the liner plate and its anchors, and the prestressing system, if any; placement of concrete; and erection tolerances of the liner plate, reinforcement, and prestressing system.

Special, new, or unique construction techniques, such as slip forming, if proposed, should be described, and the effects that these techniques may have on the structural integrity of the completed containment should be discussed.

The detailed program for the use of grouted tendons for the containment structure, if proposed, should be completely described in the PSAR and should indicate the extent to which the recommendations of Regulatory Guide 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures," are followed. At the time of submittal of the FSAR, or earlier if desirable, information on the following subjects should be included:

1. Deviations in the materials and methods from those proposed in the PSAR.

2. Grout properties from tests on the grout.

3. Test results demonstrating the suitability of the sheathing and splices, effective grouting of curved and vertical tendons, and acceptable grout under all bearing plate configurations.

143 019

3.8.1.7 Testing and Inservice Inspection Requirements. The testing and inservice inspection program for the containment should be described with emphasis on the extent of compliance with Articles CC-6000 and CC-9000 of the ASME Code, Section III, Division 2, and the extent to which the recommendations of Regulatory Guides 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments;" 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures;" and 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," are followed. Discussion of the initial structural integrity testing, as well as those tests related to the inservice inspection programs and requirements, should be provided. Information pertaining to the incorporation of inservice inspection programs into the Technical Specifications should be provided. The objectives of the tests, as well as the acceptance criteria for the results, should be defined. If new or previously untried design approaches are used, the extent of additional testing and inservice inspection should be discussed.

3.8.2 Steel Containment

This section should provide information similar to that requested in Section 3.8.1, but for steel containments and for Class MC (see ASME Code, Section III, Subsection NE) vessels, parts, or appurtenances of steel or concrete containments. In particular, the information described below should be provided.

3.8.2.1 Description of the Containment. A physical description of the steel containment and other Class MC components should be provided and supplemented with plan and section views sufficient to define the primary structural aspects and elements relied upon to perform the containment or other Class MC component function.

The geometry of the containment or component, including plan views at various elevations and sections in at least two orthogonal directions, should be provided. The arrangement of the containment shell, particularly the relationship and interaction of the shell with its surrounding shield building and with its interior compartments and floors, should be provided to establish the effect that these structures could have upon the design boundary conditions and expected behavior of the shell when subjected to the design loads.

General information related to cylindrical containment shells should include the following:

1. The foundation of the steel containment.

a. If the bottom of the steel containment is continuous through an inverted dome, the method by which this inverted dome and its supports are anchored to the concrete foundation should be described. The foundation, however, should be described in Section 3.8.5.

143 020

b. If the bottom of the steel containment is not continuous, and where a concrete base slab covered with a liner plate is used for a foundation, the method of anchorage of the steel shell cylindrical walls in the concrete base slab, particularly the connection between the floor liner plate and the steel shell, should be described. The concrete foundation, however, should be described in Section 3.8.1.

2. The cylindrical portion of the shell, including major structural attachments, such as beam seats, pipe restraints, crane brackets, and shell stiffeners, if any, in the hoop and vertical directions.

3. The dome of the steel shell, including any reinforcement at the dome/wall junction, penetrations or attachments on the inside such as supports for containment spray piping, and any stiffening of the dome.

4. Major penetrations of steel or concrete containments, or portions thereof, in particular, portions of the penetrations that are intended to resist pressure but are not backed by concrete, including sleeved and unsleeved piping penetrations, mechanical systems penetrations such as fuel transfer tubes, electrical penetrations, and access openings such as the equipment hatch and personnel locks.

Similar information should be provided for containments that are not of the cylindrical type.

<u>3.8.2.2 Applicable Codes, Standards, and Specifications</u>. This section should provide information similar to that requested in Section 3.8.1.2 for concrete containment but as applicable to steel containments or other Class MC components.

<u>3.8.2.3 Loads and Load Combinations</u>. The loads used in the design of the steel containment or other Class MC components should be specified with emphasis on the extent of compliance with Article NE-3000 of the ASME Code, Section III, Division 1, and the extent to which the recommendations of Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," are followed. The items listed below should be included.

1. Those loads encountered during preoperational testing.

2. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, hydrostatic loads such as those present in pressure-suppression containments utilizing water, and localized transient pressure loads such as those induced by actuation of safety relief valves in BWRs.

3. Those loads that would be sustained in the event of severe environmental conditions, including those that would be induced by the design wind (if the containment is not protected by a shield building) and the Operating Basis Earthquake.

4. Those loads that would be sustained in the event of extreme environmental conditions, including those that would be induced by the Design Basis Tornado (if the containment is not protected by a shield building) and the Safe Shutdown Earthquake.

5. Those loads that would be sustained in the event of abnormal plant conditions, including the design basis loss-of-coolant accident. Loads generated by other postulated accidents involving various high-energy pipe ruptures should also be discussed. Loads induced on the containment by such accidents should include associated temperature effects, pressures, and possible localized impact loads such as jet impingement and associated missile impact. Also, external pressure loads generated by events inside or outside the containment should be discussed.

6. If applicable, those loads that would be encountered, after abnormal plant conditions, including flooding of the containment subsequent to a postulated loss-of-coolant accident for the purpose of fuel recovery.

The various combinations of the above loads that should be discussed include the following: testing loads, normal operating loads, normal operating loads with severe environmental loads, normal operating loads with severe environmental loads and abnormal loads, normal operating loads with extreme environmental loads and abnormal loads, and post-LOCA flooding loads with severe environmental loads, if applicable.

Unless the steel containment is protected by a shield building, other site-related or plant-related design loads may also be applicable, as explained in Section 3.8.1.3, and should be addressed accordingly.

<u>3.8.2.4 Design and Analysis Procedures</u>. The procedures that will be used in the design and analysis of the steel containment should be described, with emphasis on the extent of compliance with Subsection NE of the ASME Code, Section III, Division 1. In particular, the following subjects should be discussed: (1) the manner in which local buckling effects are treated, (2) the expected behavior under loads, including loads that may be nonaxisymmetric and localized, and (3) the computer programs utilized. These programs should be referenced to permit identification with available published programs. Proprietary computer programs should be described in sufficient detail to establish the applicability of the programs and the measures taken to validate the programs with solutions derived from other acceptable programs or with solutions of classical problems.

<u>3.8.2.5 Structural Acceptance Criteria</u>. The acceptance criteria related to allowable stresses and other response characteristics that identify quantitatively the structural behavior of the containment should be specified with emphasis on the extent of compliance with Subsection NE of the ASME Code, Section III, Division 1, and the extent to which the recommendations of Regulatory Guide 1.57 are followed. The criteria

addressing the various loading combinations specified should be presented in terms of allowable limits for at least the following major parameters:

1. Primary stresses, including general membrane, local membrane, and bending plus local membrane stresses.

2. Primary and secondary stresses.

3. Peak stresses.

4. Buckling criteria.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques. The materials that are to be used in the construction of the steel containment should be identified with emphasis on the extent of compliance with Article NE-2000 of Subsection NE of the ASME Code, Section III, Division 1. Among the major materials that should be identified are the following:

1. Steel plates used as shell components.

2. Structural steel shapes used for stiffeners, beam seats, and crane brackets. Corrosion protection procedures should be described.

The quality control program that is proposed for the fabrication and construction of the containment should be described with emphasis on the extent of compliance with Article NE-5000 of the ASME Code, Section III, Division 1, including the following:

1. Nondestructive examination of the materials, including tests to determine their physical properties.

2. Welding procedures.

3. Erection tolerances.

Special construction techniques, if proposed, should be described, and potential effects on the structural integrity of the completed containment should be discussed.

3.8.2.7 Testing and Inservice Inspection Requirements. The testing and inservice inspection programs for the containment should be described with emphasis on the extent of compliance with Article NE-6000 of Subsection NE of the ASME Code, Section III, Division 1. A discussion of the proposed initial structural testing, including the objectives of the test and the acceptance criteria for the results, should be provided. If new or previously untried design approaches are used, the extent of additional testing and inservice inspection should be discussed. The structural integrity testing criteria for components of the containment such as personnel and equipment locks should be provided. Test program criteria for any other components that are relied upon for containment integrity should be submitted. Programs for inservice inspection in areas subject to corrosion should be provided.

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containments

This section should provide information similar to that requested in Section 3.8.1, but for internal structures of the containment. In particular, the information described below should be provided.

3.8.3.1 Description of the Internal Structures. Descriptive information, including plan and section views of the various internal structures, should be provided to define the primary structural aspects and elements relied upon to perform the safety-related functions of these structures.

General arrangement diagrams and principal features of major internal structures should be provided. Among the major structures that should be described are:

1. For PWR dry containments:

   a. Reactor support system.

   b. Steam generator support system.

   c. Reactor coolant pump support system.

   d. Primary shield wall and reactor cavity.

   e. Secondary shield walls.

   f. Other major interior structures, as appropriate, and including the pressurizer supports, the refueling pool walls, the operating floor, intermediate floors, and the polar crane supporting elements.

2. For PWR ice-condenser containments:

   a. All structures listed in item 1 above, as appropriate.

   b. The divider-barrier.

   c. The ice-condenser elements.

3. For BWR containments:*

   a. Drywell structure and appurtenances such as the drywell head and major penetrations.

143 024

b. Weir wall.

c. Refueling pool and operating floor.

d. Reactor and recirculation pump and motor support system.

e. Reactor pedestal.

f. Reactor shield wall.

g. Other major interior structures, as appropriate, including the various platforms inside and outside the drywell and the polar crane supporting elements.

3.8.3.2 Applicable Codes, Standards, and Specifications. This section should provide information similar to that requested in Section 3.8.1.2 for concrete containments, but as applicable to the internal structures of the containment as listed in Section 3.8.3.1.

3.8.3.3 Loads and Load Combinations. Among the loads used in the design of the containment internal structures listed in Section 3.8.3.1 that should be specified are the following:

1. Loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those present in refueling and pressure-suppression pools.

2. Loads that would be sustained in the event of severe environmental conditions, including those induced by the Operating Basis Earthquake.

3. Loads that would be sustained in the event of extreme environmental conditions, including those that would be induced by the Safe Shutdown Earthquake.

4. Loads that would be sustained in the event of abnormal plant conditions, including the design basis loss-of-coolant accident and other high-energy pipe rupture accidents. Loads that should be discussed include compartment pressures, jet impingement and reaction forces due to pipe rupture, elevated temperatures, impact forces of associated missiles and whipping pipes, and loads applicable to some structures such as pool swell loads in the BWR Mark III containment and drag forces in the ice-condenser PWR containment.

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* The structures listed are those of the BWR Mark III containment. For other BWR containment concepts, the applicable major interior structures should be described accordingly.

143 025

The various combinations of the above loads that should be discussed include, as a minimum, normal operating loads, normal operating loads with severe environmental loads, normal operating loads with extreme environmental loads, normal operating loads with abnormal loads, normal operating loads with severe environmental loads and abnormal loads, and normal operating loads with extreme environmental loads and abnormal loads.

In addition, the following information should be provided:

1. The extent to which the applicant's criteria comply with ACI-349, "Proposed ACI Standard: Code Requirements for Nuclear Safety Related Concrete Structures," for concrete, and with the AISC "Specification for Design, Fabrication and Erection of Structural Steel for Buildings,"* for steel, as applicable.

2. For concrete pressure-resisting portions of the divider barrier of the PWR ice-condenser containment and for concrete pressure-resisting portions of the drywell of the Mark III BWR containment, the extent to which the applicant's criteria comply with Article CC-3000 of the ASME Code, Section III, Division 2.

3. For steel pressure-resisting portions of the structures described in item 2 above, the extent to which the applicant's criteria comply with Article NE-3000 of Subsection NE of the ASME Code, Section III, Division 1, and the extent to which the recommendations of Regulatory Guide 1.57 are followed.

4. For steel linear supports of the reactor coolant system, the extent to which the applicant's criteria comply with Subsection NF of the ASME Code, Section III, Division 1.

3.8.3.4 Design and Analysis Procedures. The procedures that will be used in the design and analysis of at least those internal structures listed in Section 3.8.3.1 should be described, including the assumptions made and the identification of boundary conditions. The expected behavior under load and the mechanisms for load transfer to these structures and then to the containment base slab should be provided. Computer programs that are utilized should be referenced to permit identification with available published programs. Proprietary computer programs should be described to the maximum extent practical to establish the applicability of the programs and the measures taken to validate the programs with solutions derived from other acceptable programs or with solutions of classical problems.

The extent to which the design and analysis procedures comply with ACI-349 and with the AISC Specifications for concrete and steel structures, respectively, should be provided as applicable.

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*Copies of the AISC Specifications may be obtained from American Institute for Steel Construction, 100 Park Ave., New York, New York 10017.

For reactor coolant system linear supports, the design and analysis procedures utilized, including the type of analysis (elastic or plastic), the methods of load transfer, particularly seismic and accident loads, and the assumptions on boundary conditions, should be provided. Specifically, the extent of compliance with design and analysis procedures delineated in Subsection NF of the ASME Code, Section III, Division 1, should be indicated.

For PWR primary shield walls and BWR reactor pedestals and shield walls, the design and analysis procedures utilized should be described, including the manner by which the individual loads and load combinations are transferred to the walls and their foundations. In particular, the description should cover the normal operating thermal gradient, if any, seismic loads, and accident loads, particularly pipe rupture jet and reaction forces and cavity pressures as they may act on the entire cavity or on portions thereof.

For secondary shield walls and operating and intermediate floors, the design and analysis procedures utilized for these walls and floors, including assumptions on structural framing and behavior under loads, should be described. Where elastoplastic behavior is assumed and the ductility of the walls is relied upon to absorb the energy associated with jet and missile loads, the procedures and assumptions should be described with particular emphasis on modeling techniques, boundary conditions, force-time functions, and assumed ductility. For the differential pressure, methods of ensuring elastic behavior should be described, particularly in determining an equivalent static load for the impulsive pressure load.

For concrete pressure-resisting portions of the divider barrier of the PWR ice-condenser containment and for concrete pressure-resisting portions of the drywell of the BWR Mark III containment, the extent to which the applicant's criteria comply with Article CC-3000 of the ASME Code, Section III, Division 2, should be provided. For steel pressure-resisting portions of these two structures, discuss the extent to which the applicant's criteria comply with Article NE-3000 of Subsection NE of the ASME Code, Section III, Division 1, and the extent to which the recommendations of Regulatory Guide 1.57 are followed.

<u>3.8.3.5 Structural Acceptance Criteria</u>. This section should provide information similar to that requested in Section 3.8.1.5 for concrete containments, but as applicable to the various containment internal structures listed in Section 3.8.3.1.

For each applicable load combination listed in Section 3.8.3.3, the allowable limits should be provided, as applicable, for stresses, strains, deformation (particularly for the RCS linear supports), and factors of safety against structural failure. The extent of compliance with the various applicable codes, as indicated in Section 3.8.3.3, should be presented.

143 027

3.8.3.6 Materials, Quality Control, and Special Construction Techniques. The materials, quality control programs, and any special construction techniques should be identified and described.

Among the major materials of construction that should be described are the concrete ingredients, the reinforcing bars and splices, and the structural steel and various supports and anchors.

The quality control program proposed for the fabrication and construction of the containment interior structures should be described, including nondestructive examination of the materials to determine physical properties, placement of concrete, and erection tolerances.

Special, new, or unique construction techniques should be described to determine their effects on the structural integrity of the completed interior structure.

In addition, the following information should be provided:

1. The extent to which the material and quality control requirements comply with ACI-349 for concrete, and with the AISC Specifications for steel, as applicable.

2. For steel linear supports of the reactor coolant system, the extent to which the material and quality control requirements comply with Subsection NF of the ASME Code, Section III, Division 1.

3. For quality control in general, the extent to which the applicant complies with ANSI N45.2.5, and follows the recommendations of Regulatory Guide 1.55, "Concrete Placement in Category I Structures."

4. If welding of reinforcing bars is proposed, the extent to which the design complies with the ASME Code, Section III, Division 2. Any exceptions taken should be identified and justified.

3.8.3.7 Testing and Inservice Inspection Requirements. The testing and inservice inspection programs for the internal structures should be described. Test requirements for internal structures related directly and critically to the functioning of the containment concept such as the drywell of the BWR Mark III containment should be specified. Inservice inspection requirements, when needed, should also be described. The extent of compliance with the applicable codes as described in Section 3.8.3.6 should be indicated.

3.8.4 Other Seismic Category I Structures

Information should be provided in this section for all Seismic Category I structures not covered by Sections 3.8.1, 3.8.2, 3.8.3, or 3.8.5. The information should be similar to that requested for Section 3.8.1. In particular, the information described below should be provided.

143 028

3.8.4.1 Description of the Structures. Descriptive information, including plan and section views of each structure, should be provided to define the primary structural aspects and elements relied upon for the structure to perform its safety-related function. The relationship between adjacent structures, including any separation or structural ties, should be described. Among the plant Seismic Category I structures that should be described are the following:

1. Containment enclosure buildings.

2. Auxiliary buildings.

3. Fuel storage buildings.

4. Control buildings.

5. Diesel generator buildings.

6. Other Seismic Category I structures, as applicable, including such structures as pipe and electrical conduit tunnels, waste storage facilities, stacks, intake structures, pumping stations, water wells, cooling towers, and concrete dams, embankments, and tunnels. Structures that are safety related but because of other design provisions are not classified as Seismic Category I should also be described.

3.8.4.2 Applicable Codes, Standards, and Specifications. Information similar to that requested in Section 3.8.1.2 for concrete containments, but as applicable to all other Seismic Category I structures, should be provided.

3.8.4.3 Loads and Load Combinations. The loads used in the design of all other Seismic Category I structures should be specified, including:

1. Those loads encountered during normal plant startup, operation, and shutdown, including dead loads, live loads, thermal loads due to operating temperature, and hydrostatic loads such as those in spent fuel pools.

2. Those loads that would be sustained in the event of severe environmental conditions, including those that would be induced by the Operating Basis Earthquake (OBE) and the design wind specified for the plant site.

3. Those loads that would be sustained in the event of extreme environmental conditions, including those that would be induced by the Safe Shutdown Earthquake (SSE) and the Design Basis Tornado specified for the plant site.

4. Those loads that would be sustained in the event of abnormal plant conditions. Such abnormal plant conditions include the postulated rupture of high-energy piping. Loads induced by such an accident include

143 029

elevated temperatures and pressures within or across compartments and possibly jet impingement and impact forces usually associated with such ruptures.

The various combinations of the above loads that should be discussed include normal operating loads, normal operating loads with severe environmental loads, normal operating loads with extreme environmental loads, normal operating loads with abnormal loads, normal operating loads with severe environmental loads and abnormal loads, and normal operating loads with extreme environmental loads and abnormal loads.

The loads and load combinations described above are generally applicable to most structures. However, other site-related design loads might also be applicable. Such loads include those induced by floods, potential aircraft crashes, explosive hazards in proximity to the site, and projectiles and missiles generated from activities of nearby military installations.

3.8.4.4 Design and Analysis Procedures. The design and analysis procedures should be described with emphasis on the extent of compliance with ACI-349 and the AISC Specifications for concrete and steel structures, respectively, including the assumptions made on boundary conditions. The expected behavior under load and the mechanisms of load transfer to the foundations should be provided. Computer programs should be referenced to permit identification with available published programs. Proprietary computer programs should be described to the maximum extent practical to establish the applicability of the program and the measures taken to validate the program with solutions derived from other acceptable programs or with solutions of classical problems.

3.8.4.5 Structural Acceptance Criteria. The design criteria relating to stresses, strains, gross deformations, factors of safety, and other parameters that identify quantitatively the margins of safety should be specified with emphasis on the extent of compliance with ACI-349 for concrete and with the AISC Specifications for steel.

3.8.4.6 Materials, Quality Control, and Special Construction Techniques. The materials, quality control programs, and any new or special construction techniques should be addressed as outlined in Section 3.8.3.6.

3.8.4.7 Testing and Inservice Inspection Requirements. The testing and inservice inspection requirements, if any, should be specified.

3.8.5 Foundations

The information provided in this section should be similar to that requested under Section 3.8.1 for concrete containments but as applicable to foundations of all Seismic Category I structures. Concrete foundations of steel or concrete containments should be discussed in Section 3.8.1 and in this section appropriate.

143 030

The information should address foundations for all Seismic Category I structures constructed of materials other than soil for the purpose of transferring loads and forces to the basic supporting media. In particular, the information described below should be provided.

3.8.5.1 Description of the Foundations. Descriptive information, including plan and section views of each foundation, should be provided to define the primary structural aspects and elements relied upon to perform the foundation function. The relationship between adjacent foundations, including any separation provided and the reasons for such separation, should be described. In particular, the type of foundation and its structural characteristics should be discussed. General arrangement of each foundation should be provided with emphasis on the methods of transferring horizontal shears, such as those seismically induced, to the foundation media. If shear keys are utilized for such purposes, the general arrangement of the keys should be included. If waterproofing membranes are utilized, their effect on the capability of the foundation to transfer shears should be discussed.

Information should be provided to adequately describe other types of foundation structures such as pile foundations, caisson foundations, retaining walls, abutments, and rock and soil anchorage systems.

3.8.5.2 Applicable Codes, Standards, and Specifications. Information similar to that requested in Section 3.8.1.2, but as applicable to foundations of all Seismic Category I structures, should be provided.

3.8.5.3 Loads and Load Combinations. This section should provide similar information to that requested in Section 3.8.4.3, but as applicable to the foundations of all Seismic Category I structures.

3.8.5.4 Design and Analysis Procedures. This section should provide information applicable to the foundations of all Seismic Category I structures. The information should be similar to that requested in Section 3.8.4.4.

In particular, the assumptions made on boundary conditions and the methods by which lateral loads and forces and overturning moments thereof are transmitted from the structure to the foundation media should be discussed, and the methods by which the effects of settlement are taken into consideration should be described.

3.8.5.5 Structural Acceptance Criteria. This section should provide information applicable to foundations of all Seismic Category I structures. The information should be similar to that requested in Section 3.8.4.5.

In particular, the design limits imposed on the various parameters that serve to define the structural stability of ar' structure and its foundations should be indicated, including differential settlements and factors of safety against overturning and sliding.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques. This section should provide information for the foundations of all Seismic Category I structures. The information should be similar to that requested in Section 3.8.4.6.

3.8.5.7 Testing and Inservice Inspection Requirements. This section should discuss information for the foundations of all Seismic Category I structures. The information should be similar to that requested in Section 3.8.4.7.

If programs for continued surveillance and monitoring of foundations are required, a discussion to define the various aspects of the program should be provided.

3.9 Mechanical Systems* and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients. Provide a complete list of transients to be used in the design and fatigue analysis of all ASME Code Class 1 and CS components, component supports, and reactor internals. The number of events for each transient should be included, along with assurance that the number of load and stress cycles per event is properly taken into account. All design transients that are contained in the ASME Code-required "Design Specifications" for the components of the reactor coolant pressure boundary should be specified. Examples of such transients are startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients resulting from single operator errors, inservice hydrostatic tests, and seismic events. All transients or combinations of transients should be classified with respect to the plant and system operating condition categories identified as "normal," "upset," "emergency," "faulted," or "testing."

3.9.1.2 Computer Programs Used in Analyses. Provide a list of computer programs that will be used in dynamic and static analyses to determine structural and functional integrity of all Seismic Category I systems, components, equipment, and supports. Include a brief description of each program, the extent of its application, and the design control measures, required per Appendix B of 10 CFR Part 50, that will be employed to demonstrate the applicability and validity of each program.

3.9.1.3 Experimental Stress Analysis. If experimental stress analysis methods are used in lieu of analytical methods for Seismic Category I

*Fuel system design information is addressed in Section 4.2.

143 032

ASME Code and non-Code items, sufficient information should be provided to show the validity of the design.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition. The analytical methods (e.g., elastic or elastic-plastic) used to evaluate stresses for Seismic Category I ASME Code and non-Code items should be described, including a discussion of their compatibility with the type of dynamic system analysis used. The stress-strain relationship and ultimate strength used in the analysis for each component should be shown to be valid. If the use of elastic, elastic-plastic, or limit item analysis concurrently with elastic or elastic-plastic system analysis is invoked, the basis for these procedures should provide assurance that the calculated item or item support deformations and displacements do not violate the corresponding limits and assumptions on which the method used for the system analysis is based. When elastic-plastic stress or deformation design limits are specified for ASME Code and non-Code items, the methods of analysis used to calculate the stresses and/or deformations resulting from the faulted condition loadings should be provided. Describe the procedure for developing the loading function on each component.

3.9.2 Dynamic Testing and Analysis

The criteria, testing procedures, and dynamic analyses employed to ensure structural and functional integrity of piping systems, mechanical equipment, and reactor internals under vibratory loadings, including those due to fluid flow and postulated seismic events, should be provided.

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects. Information should be provided concerning the piping vibration, thermal expansion, and dynamic effects testing that will be conducted during startup functional testing on (1) ASME Code Class 1, 2, and 3 systems, (2) other high-energy piping systems inside Seismic Category I structures, (3) high-energy portions of systems whose failure could reduce the functioning of any Seismic Category I plant feature to an unacceptable level, and (4) Seismic Category I portions of moderate-energy piping systems located outside containment. The purpose of these tests is to confirm that these piping systems, restraints, components, and supports have been designed adequately to withstand the flow-induced dynamic loadings under operational transient and steady-state conditions anticipated during service. In addition, the thermal motions should be monitored to ensure that adequate clearances are provided to allow unrestrained normal thermal movement of systems, components, and supports. The program should include a list of different flow modes, a list of selected locations for visual inspection and measurements, the acceptance criteria, and the possible corrective actions if excessive vibration occurs.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment. Seismic qualification testing of safety-related mechanical equipment is required to ensure its functional integrity and operability

143 033

during and after a postulated seismic occurrence. The following information should be provided in the PSAR:

1. The criteria for seismic qualification, such as the deciding factors for choosing test and/or analysis, considerations in defining the input motion at the equipment monitoring locations, and the process to demonstrate adequacy of the seismic qualification program.

2. The methods and procedures used to test Seismic Category I mechanical equipment operation during and after the Safe Shutdown Earthquake (SSE) and to ensure structural and functional integrity of the equipment after several occurrences of the Operating Basis Earthquake (OBE) in combination with normal operating loads. Included are mechanical equipment such as fans, pump drives, heat exchanger tube bundles, valve actuators, battery and instrument racks, control consoles, cabinets, panels, and cable trays. Broad-band seismic excitation, dynamic coupling, and multidirectional loading effects should be considered in the development of the seismic qualification program.

3. The methods and procedures of analysis and for testing of the supports for the above Seismic Category I mechanical equipment, and the verification procedures used to account for the possible amplification of design loads (amplitude and frequency content) under seismic conditions.

There should be provided in the FSAR the results of tests and analyses to ensure the proper implementation of the criteria accepted in the construction permit (CP) review and to demonstrate adequate seismic qualification.

<u>3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions</u>. A description of the dynamic system analysis of structural components within the reactor vessel caused by the operational flow transients and steady-state conditions should be provided in the PSAR. The purpose of this analysis is to demonstrate the acceptability of the reactor internals design for normal operating conditions and to predict the input forcing functions and the vibratory response of the reactor internals prior to conducting the preoperational vibration test of a prototype reactor. Information concerning the method of analysis, the specific locations for response calculation, the considerations to define the mathematical model, and the acceptance criteria should be provided in the PSAR.

<u>3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals</u>. Information should be provided in the PSAR describing the extent to which the recommendations of preoperational flow-induced vibration testing of reactor internals during the startup functional test program, as delineated in Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," will be implemented. The purpose of this test is to

143 034

demonstrate that flow-induced vibrations experienced during normal operation will not cause structural failure or degradation. For the prototype reactor, information in the PSAR should include a list of flow modes, a list of sensor types and locations, a description of test procedures, methods used to process and interpret the measured data, and the procedures for implementing the visual inspection. For a reactor internal with the same design, size, configuration, and operation conditions as an identified valid prototype reactor internal, indicate the extent to which the preoperational vibration test program follows the recommendations for non-prototype testing presented in Regulatory Guide 1.20; provide justification for any alternative approach.

3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Condition. The following information should be included in the discussion of the dynamic system analysis methods and procedures used to confirm the structural design adequacy of the reactor internals and the unbroken loop of the reactor piping system to withstand dynamic effects with no loss of function under a simultaneous occurrence of LOCA or steam line break and Safe Shutdown Earthquake (SSE):

1. Typical diagrams of the dynamic system mathematical modeling of piping, pipe supports, and reactor internals, along with fuel element assemblies and control rod assemblies and drives, that will be used in the analysis, including a discussion of the bases for any structural partitioning and directional decoupling of components (PSAR).

2. A description of the methods used to obtain the forcing functions and a description of the forcing functions that will be used for the dynamic analysis of the LOCA or steam line break and SSE event, including system pressure differentials, direction, rise time, magnitude, duration, initial conditions, spatial distribution, and loading combinations (PSAR).

3. A description of the methods and procedures that will be used to compute the total dynamic structural responses, including the buckling response, of those structures in compression (PSAR).

4. A summary of the results of the dynamic analysis (FSAR).

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results (FSAR). A discussion should be provided that describes the method to be used for correlating the results from the reactor internals preoperational vibration test with the analytical results derived from dynamic analyses of reactor internals under operational flow transients and steady-state conditions. In addition, this discussion may include procedures for verifying the mathematical model used in the faulted condition (LOCA, steam line break, and SSE) by comparing certain dynamic characteristics such as natural frequencies.

143 035

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The information requested in Sections 3.9.3.1 through 3.9.3.4 should be provided for components and component supports constructed in accordance with Division 1 of Section III of the ASME Code. Section 3.9.3 includes ASME Code Class 1, 2, and 3 components, core support (CS) structures, and component supports; Class MC is covered in Section 3.8.3. The design information relative to component design for steam generators as called for in Section 5.4.2 should be incorporated in this section. This includes field run piping and internal parts of components.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits. Provide the combination of loading conditions and the design transients applicable to the design of each ASME Code constructed item for each system. Identify for each initiating event (i.e., LOCA, SSE, pipe break, and other transients) the appropriate plant operating condition and the appropriate component operating condition used to establish the design stress limits for the ASME Code constructed items (see Section 3.9.1.1).

The actual design condition (including test condition) stress limits and deformation criteria selected for design (for the combination of loading conditions and design transients established as described above) should be presented. Design stress limits that allow inelastic deformation (comparable to faulted condition design limits) should be identified, and a description of the procedures that will be used for analysis or test should be provided in the PSAR (see Section 3.9.1.4).

The FSAR should include the following for ASME Code Class 1 components, CS structures, and ASME Code Class 1 component supports:

1. A summary description of mathematical or test models used,

2. Methods of calculation or test, including simplifying assumptions, identification of method of system and component analysis used, and demonstration of their compatibility (see Section 3.9.1.4) in the case of components and supports designed to faulted limits, and

3. A summary of the maximum total stress, deformation, and cumulative usage factor values should be provided in the FSAR for each of the component operating conditions for all ASME Code Class 1 components. Identify those values that differ from the allowable limits by less than 10%, and provide the contribution of each of the loading categories, such as seismic, dead weight, pressure, and thermal, to the total stress for each maximum stress value identified in this range.

The FSAR should include the following for all other classes of components and their supports:

143 036

1. A summary description of any test models used (see Section 3.9.1.´ .

2. A summary description of mathematical models or test models used to evaluate the faulted conditions, as appropriate, for components and supports (see Sections 3.9.1.2 and 3.9.1.4).

3. For all ASME Code Class 2 and 3 components required to shut down the reactor or mitigate the consequences of a postulated piping failure without offsite power, a summary of the maximum total stress and deformation values should be provided in the FSAR for each of the component operating conditions. Identify those values that differ from the allowable limits by less than 10%.

The PSAR should include a listing of transients appropriate to ASME Code Class 1, 2, and 3 components, CS structures, and component supports and should be categorized on the basis of plant operating condition. In addition, for ASME Code Class 1 components and CS and ASME Code Class 1 component supports, include the number of cycles to be used in the fatigue analysis appropriate to each transient (see Section 3.9.1.1).

3.9.3.2 Pump and Valve Operability Assurance. Provide a list that identifies all active ASME Class 1, 2, and 3 pumps and valves. Present the criteria to be employed in a test program, or program consisting of test and analysis, to ensure the operability of pumps required to function and valves required to open or close to perform a safety function during or following the specified plant event. Discuss the features of the program, and include conditions of test, scale effect if appropriate, loadings for specified plant event, transient loads, including seismic component, dynamic coupling to other systems, stress limits, deformation limits, and other information considered pertinent to assurance of operability. Design stress limits established as provided for in Section 3.9.3.1 should be included in the program. All of the above should be included in the PSAR.

The FSAR should include program results summarizing stress and deformation levels and environmental qualification, as well as maximum test envelope conditions for which the component qualifies, including end connection loads and operability results.

3.9.3.3 Design and Installation Details for Mounting of Pressure-Relief Devices. The design and installation criteria applicable to the mounting of the pressure-relieving devices (safety valves and relief valves) for the overpressure protection of ASME Class 1 and 2 system components should be described. Information pertaining to loading combinations should identify the most severe combination of the applicable loads due to internal fluid pressure, fluid states, dead weight of valves and piping, thermal load under heatup, steady-state and transient valve operation, reaction forces when valves are discharging (valve opening sequence and opening

143 037

times), and seismic events (i.e., Operating Basis Earthquake and Safe Shutdown Earthquake).

The method of analysis and magnitude of any dynamic load factors used should be included. Discharge piping effects (i.e., closed or open system) should be discussed and included in the analysis. The PSAR should include the criteria presented above, and the FSAR should present the results of the analysis.

3.9.3.4 Component Supports. Loading combinations, design transients, stress limits, and deformation limits should be provided as discussed in Section 3.9.3.1.

The supports for active components should be tested, analyzed, or analyzed and tested, as discussed for components in Section 3.9.3.2, and their effects on operability included in the discussion provided in that section.

The PSAR should present the criteria to be used, and the FSAR should present the results of analysis or test programs as discussed in Sections 3.9.3.1 and 3.9.3.2.

3.9.4 Control Rod Drive Systems

Information on the control rod drive systems (CRDS) should be provided by the applicant in the SAR for review by the staff. For electromagnetic systems, this includes the control rod drive mechanism (CRDM) and extends to the coupling interface with the reactivity control elements. For hydraulic systems, this includes the CRDM, the hydraulic control unit, the condensate supply system, and the scram discharge volume and extends to the coupling interface with the reactivity control elements. For both types of systems, the CRDM housing should be treated as part of the reactor coolant pressure boundary (RCPB). Information on CRDS materials should be included in Section 4.5.1.

If other types of CRDS are proposed or if new features that are not specifically mentioned here are incorporated in current types of CRDS, information should be supplied for the new systems or new features.

3.9.4.1 Descriptive Information of CRDS. The descriptive information, including design criteria, testing programs, drawings, and a summary of the method of operation of the control rod drives, should be provided to permit an evaluation of the adequacy of the system to properly perform its design function.

3.9.4.2 Applicable CRDS Design Specifications. Information should be provided pertaining to design codes, standards, specifications, and standard practices, as well as to NRC general design criteria, regulatory guides, and positions that are applied in the design, fabrication, construction, and operation of the CRDS.

143 038

The various criteria should be supplied along with the names of the apparatus to which they apply. Pressurized parts of the system should be listed or referenced in Section 3.2.2 in order to determine the extent to which the applicant complies with the Class 1 requirements of Section III of the ASME Code for those portions that are part of the reactor coolant pressure boundary, and with other specified parts of Section III or other sections of the ASME Code for pressurized portions that are not part of the reactor coolant pressure boundary.

Information should be provided to evaluate the nonpressurized portions of the control rod drive system to determine the acceptability of design margins for allowable values of stress, deformation, and fatigue used in the analyses. If an experimental testing program is used in lieu of analysis, the program should be provided. The program description should adequately cover the areas of concern in the determination and verification of the stress, deformation, and fatigue in the CRDS.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations. Information should be presented that pertains to the applicable design loads and their appropriate combinations, to the corresponding design stress limits, and to the corresponding allowable deformations. The deformations are of interest in the present context only in those instances where a failure of movement could be postulated to occur and such movement would be necessary for a safety-related function.

If the applicant selects an experimental testing option in lieu of establishing a set of stress allowables and deformation allowables, an extensive description of the testing program should be provided. The load combination, design stress limit, and allowable deformation criteria should be provided in the PSAR.

The design limits and safety margins for those components not designed to the ASME Code should be specified in the FSAR, or alternatively a commitment to provide this information prior to fuel loading should be made in the FSAR. Information similar to that requested in Section 3.9.3 should be provided for those components designed to the ASME Code.

3.9.4.4 CRDS Performance Assurance Program. Plans for the conduct of a performance assurance program or plans that reference previous test programs or standard industry procedures for similar apparatus should be provided. For example, the life cycle test program for the CRDS should be presented. The design performance assurance program presented should cover the following:

1. Life cycle test program,
2. Proper service environment imposed during test,
3. Mechanism functional tests, and
4. Program results (FSAR).

143 039

3.9.5 Reactor Pressure Vessel Internals

The information requested in Sections 3.9.5.1 through 3.9.5.3 should be provided as part of an evaluation program to ensure the structural and functional integrity of the reactor internals (includes ASME Class CS (core supports) and non-ASME-Code-covered internals). Information on reactor internals materials should be included in Section 4.5.2.

3.9.5.1 Design Arrangements. The physical or design arrangements of all reactor internals structures, components, assemblies, and systems should be presented, including the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects. The functional requirements for each component should be described. Verify that any significant changes in design from those in previously licensed plants of similar design do not affect the flow-induced vibration test results requested in Section 3.9.2.

3.9.5.2 Loading Conditions. The plant and system operating conditions and design basis events that provide the basi for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events should be specified in accordance with the information requested in Section 3.9.1.1.

3.9.5.3 Design Bases. Provide the specific design and service loading combinations applicable to reactor internals. These loading combinations should consider all the conditions in 3.9.5.2 above. Describe the method of combination of these loads. For each specific loading combination, provide the design or service limits to be applied to the reactor internals. Provide the deflection, cycling, and fatigue limits. Verify that the allowable deflections will not interfere with the functioning of all related components (e.g., control rods and standby cooling systems). Indicate the extent to which the design and construction of the core support structures is in accordance with Subsection NG of the ASME Code. Indicate the extent to which the design of other reactor internals will be consistent with Article NG-3000. A summary of the maximum calculated total stress, deformation, and cumulative usage factor should be provided in the FSAR for each designated design or service limit. Details of the dynamic analyses should be presented in Section 3.9.2 of the SAR.

3.9.6 Inservice Testing of Pumps and Valves

A test program should be provided that includes baseline preservice testing and a periodic inservice test program to ensure that all ASME Code Class 1, 2, and 3 pumps provided with an emergency power source and all ASME Code Class 1, 2, and 3 valves will be in a state of operational readiness to perform their safety function throughout the life of the plant.

3.9.6.1 Inservice Testing of Pumps. Descriptive information in the PSAR should cover the inservice test program of all ASME Code Class 1, 2, and 3 system pumps provided with an emergency power source. Reference value* tests for speed, pressure, flow rate, vibration, lubrication, and bearing temperature at normal pump operating conditions should be presented. Methods for measuring the reference values and inservice values for the pump parameters listed above should be presented. In addition, the pump test plan and schedule should be provided and included in the technical specifications.

3.9.6.2 Inservice Testing of Valves. Descriptive information in the PSAR should cover the inservice test program of all ASME Code Class 1, 2, and 3 valves. The test program should include preservice tests, valve replacement, valve repair and maintenance, indication of valve position, and inservice tests for all valve categories (as defined in IWV-2100 of the ASME Code). In addition, the valve test procedure and schedule should be provided and included in the technical specifications.

3.9.6.3 Relief Requests. Paragraph 50.55a(g) of 10 CFR requires a nuclear power facility to periodically update its inservice testing program to meet the requirements of future revisions of Section XI of the ASME Code. However, if it proves impractical to implement these criteria, the applicant is allowed to submit requests for relief from Section XI requirements on a case-by-case basis. Information provided should describe the specific area of relief requested, explain why compliance with Section XI in this case is impractical, and describe any alternative test procedures.

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

All Seismic Category I instrumentation, electrical equipment, and their supports should be identified. The seismic qualification criteria applicable to the reactor protection system, engineered-safety-feature Class 1E equipment, the emergency power system, and all auxiliary safety-related systems and supports should be provided. Methods and procedures used to qualify electrical equipment, instrumentation, and their supports should also be provided.

3.10.1 Seismic Qualification Criteria

The criteria for seismic qualification, including the decision criteria for selecting a particular test or method of analysis, the considerations defining the input motion, and the process to demonstrate adequacy of the seismic qualification program, should be provided. The extent to which guidance contained in Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," will be used should be indicated.

*Defined in IWP-3112 of the ASME Code.

143 041

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

The methods and procedures used to qualify by test or analysis Seismic Category I instrumentation and electrical equipment for operation during and after the Safe Shutdown Earthquake and to ensure structural and functional integrity of the equipment after several occurrences of the Operating Basis Earthquake should be provided. Seismic Category I instrumentation and electrical equipment include the reactor protection system, engineered-safety-feature Class 1E equipment, emergency power system, and all auxiliary safety-related systems.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

The methods and procedures for analysis or testing of Seismic Category I instrumentation and electrical equipment supports and the verification procedures used to account for the possible amplification of design loads (amplitude and frequency content) under seismic conditions should be provided. Supports include items such as battery racks, instrument racks, control consoles, cabinets, panels, and cable trays.

3.10.4 Operating License Review (FSAR)

The results of tests and analyses that ensure the proper implementation of the criteria established in the construction permit review, and that demonstrate adequate seismic qualification, should be provided in the FSAR.

3.11 Environmental Design of Mechanical and Electrical Equipment

The purpose of this section is to provide information on the environmental conditions and design bases for which the mechanical, instrumentation, and electrical portions of the engineered safety features and reactor protection systems are designed to ensure acceptable performance in all environments (e.g., normal, tests, and accident).

The following specific information should be included concerning the design bases related to the capability of the mechanical, instrumentation, and electrical portions of the engineered safety features, and reactor protection system to perform their intended functions in the combined postaccident environment of temperature, pressure, humidity, chemistry, and radiation.

3.11.1 Equipment Identification and Environmental Conditions

All safety-related equipment and components (e.g., motors, cables, filters, pump seals, shielding) located in the primary containment and elsewhere that are required to function during and subsequent to any of the design basis accidents should be identified and their locations specified. For equipment inside containment, the location should specify

1 3 042

whether inside or outside the missile shield (for PWRs) or whether inside or outside the drywell (for BWRs).

Both the normal and accident environmental conditions should be explicitly defined for each item of equipment. These definitions should include the following parameters: temperature, pressure, relative humidity, radiation, chemicals, and vibration (nonseismic).

For the normal environment, including that due to loss of environmental control systems, specific values should be provided. For the accident environment, these parameters should be presented as functions of time, and the cause of the postulated environment (loss-of-coolant accident, steam line break, or other) should be identified.

The length of time that each item of equipment is required to operate in the accident environment should be provided.

3.11.2 Qualification Tests and Analyses

A description should be provided of the qualification tests and analyses that have been or will be performed on each of these items to ensure that it will perform in the combined temperature, pressure, humidity, chemical, and radiation environment. The specific values of temperature, pressure, humidity, chemicals, and radiation should be included.

Indicate how the requirements of General Design Criteria 1, 4, 23, and 50 of Appendix A to 10 CFR Part 50 and Criterion III of Appendix B to 10 CFR Part 50 will be met. The extent to which the guidance contained in the regulatory guides listed below will be utilized should be indicated:

Regulatory Guide 1.30 (Safety Guide 30), "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment;"

Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants;"

Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants;"

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

Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants;" and

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

3.11.3 Qualification Test Results

The results of the qualification tests for each type of equipment should be provided in the FSAR or should be referenced if previously submitted.

3.11.4 Loss of Ventilation

Provide the bases that ensure that loss of the air conditioning or ventilation system will not adversely affect the operability of safety-related control and electrical equipment located in the control room and other areas. The analyses performed to identify the worst case environment (e.g., temperature, humidity) should be described, including identification and determination of the limiting condition with regard to temperature that would require reactor shutdown. Any testing (factory or onsite) that has been or will be performed to confirm satisfactory operability of control and electrical equipment under extreme environmental conditions should be described. The documentation of the successful completion of qualification tests for each type of equipment should be specified in the PSAR and supplied in the FSAR.

3.11.5 Estimated Chemical and Radiation Environment

For each engineered safety feature (ESF), the design source term for the chemical and radiation environment both for normal operation and for the design basis accident environment should be identified. For engineered safety features inside containment, the chemical composition and resulting pH of the liquids in the reactor core and in the containment sump should be identified. Indicate the extent to which estimates of radiation exposures are based on a radiation source term that is consistent with Regulatory Guides 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," and 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident." Wherever alternative approaches are used, demonstrate that an acceptable level of safety has been attained. Determinations of the exposure of organic components on ESF systems should consider both beta and gamma radiation. Beta and gamma exposures should be tabulated separately and should list the average energy of each type of radiation. For ESF systems outside containment, the radiation estimates should take into account factors affecting the source term such as containment leak rate, meteorological dispersion (if appropriate), and operation of other ESF systems. The engineered safety features considered and the corresponding source terms and chemical environments should be presented in tabular form. All assumptions used in the calculation should be listed.

4. REACTOR

In this chapter of the SAR, the applicant should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operational modes, including both transient and steady state, and accident conditions. This chapter should also include information to support the analyses presented in Chapter 15, "Accident Analyses."

4.1 Summary Description

A summary description of the mechanical, nuclear, and thermal and hydraulic designs of the various reactor components, including the fuel, reactor vessel internals, and reactivity control systems, should be given. The description should indicate the independent and interrelated performance and safety functions of each component. Information on control rod drive systems and reactor vessel internals presented in Sections 3.9.4 and 3.9.5 may be incorporated by reference. A summary table of the important design and performance characteristics should be included. A tabulation of analysis techniques used and load conditions considered, including computer code names, should also be included.

4.2 Fuel System Design

The fuel system is defined as consisting of guide tubes or thimbles; fuel rods with fuel pellets, insulator pellets, cladding, springs, end closures, fill gas, and getters; water rods; burnable poison rods; spacer grids and springs; assembly end fittings and springs; channel boxes; and the reactivity control assembly. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism. The design bases for the mechanical, chemical, and thermal design of the fuel system that can affect or limit the safe, reliable operation of the plant should be presented. The description of the fuel system mechanical design should include the following aspects: (1) mechanical design limits such as those for allowable stresses, deflection, cycling, and fatigue, (2) capacity for fuel fission gas inventory and pressure, (3) a listing of material properties, and (4) considerations for radiation damage, cladding collapse time, materials selection, and normal operational vibration. Details for seismic loadings should be presented in Section 3.7.3; shock (LOCA) loadings and the effects of combined shock and seismic loads should be presented in this section. The chemical design should consider all possible fuel-cladding-coolant interactions. The description of the thermal design should include such items as maximum fuel and cladding temperatures, clad-to-fuel gap conductance as a function of burnup and operating conditions, and fuel cladding integrity criteria.

4.2.1 Design Bases

The applicant should explain and substantiate the selection of design bases from the viewpoint of safety considerations. Where the limits selected

143 045

are consistent with proven practice, a referenced statement to that effect will suffice; where the limits extend beyond present practice, an evaluation and an explanation based on developmental work or analysis should be provided. These bases may be expressed as explicit numbers or as general conditions.

The discussion of design bases should include a description of the functional characteristics in terms of desired performance under stated conditions. This should relate systems, components, and materials performance under normal operating, anticipated transient, and accident conditions. The discussion should consider the following with respect to performance:

1. Cladding

a. The mechanical properties of the cladding, e.g., Young's modulus, Poisson's ratio, design dimensions, strength, ductility, and creep rupture limits, and the effects of design temperature and irradiation on the properties,

b. Stress-strain limits,

c. Vibration and fatigue,

d. The chemical properties of the cladding.

2. Fuel Material

a. The thermal-physical properties of the fuel, e.g., melting point, thermal conductivity, density, and specific heat, and the effects of design temperature and irradiation on the properties,

b. The effects of fuel densification and fission product swelling,

c. The chemical properties of the fuel.

3. Fuel Rod Performance

a. Analytical models and the conservatism in the input data,

b. The ability of the models to predict experimental or operating characteristics,

c. The standard deviation or statistical uncertainty associated with the correlations or analytical models.

4. Spacer Grid and Channel Boxes

a. Mechanical, chemical, thermal, and irradiation properties of the materials,

143 046

b. Vibration and fatigue,

c. Chemical compatibility with other core components, including coolant.

5. Fuel Assembly

a. Structural design,

b. Thermal-hydraulic design.

6. Reactivity Control Assembly and Burnable Poison Rods

a. The thermal-physical properties of the absorber material,

b. The compatibility of the absorber and cladding materials,

c. Cladding stress-strain limits,

d. Irradiation behavior of absorber material.

7. Surveillance Program

a. The requirements for surveillance and testing of irradiated fuel rods, burnable poison rods, control rods, channel boxes, and instrument tube/thimbles.

4.2.2 Description and Design Drawings

A description and preliminary (PSAR) or final (FSAR) design drawing of the fuel rod components, burnable poison rods, fuel assemblies, and reactivity control assemblies showing arrangement, dimensions, critical tolerances, sealing and handling features, methods of support, internal pressurization, fission gas spaces, burnable poison content, and internal components should be provided. A discussion of design features that prevent improper orientation or placement of fuel rods or assemblies within the core should be included.

4.2.3 Design Evaluation

An evaluation of the fuel system design should be presented for the physically feasible combinations of chemical, thermal, irradiation, mechanical, and hydraulic interaction. Evaluation of these interactions should include the effects of normal reactor operations, anticipated transients without scram, and postulated accidents. The fuel system design evaluation should include the following:

1. Cladding

a. Vibration analysis,

143 047

b. Fuel element internal and external pressure and cladding stresses during normal and accident conditions with particular emphasis on temperature transients or depressurization accidents,

c. Potential for chemical reaction, including hydriding, fission product attack, and crud deposition,

d. Fretting and crevice corrosion,

e. Stress-accelerated corrosion,

f. Cycling and fatigue,

g. Material wastage due to mass transfer,

h. Rod bowing due to thermal, irradiation, and creep dimensional changes,

i. Consequences of power-coolant mismatch,

j. Irradiation stability of the cladding,

k. Creep collapse and creepdown.

2. Fuel

a. Dimensional stability of the fuel,

b. Potential for chemical interaction, including possible waterlogging rupture,

c. Thermal stability of the fuel, including densification, phase changes, and thermal expansion,

d. Irradiation stability of the fuel, including fission product swelling and fission gas release.

3. Fuel Rod Performance

a. Fuel-cladding mechanical interaction,

b. Failure and burnup experience, including the thermal conditions for which the experience was obtained for a given type of fuel and the results of long-term irradiation testing of production fuel and test specimens,

c. Fuel and cladding temperatures, both local and gross, with an indication of the correlation used for thermal conductivity, gap conductance as a function of burnup and power level, and the method of employing peaking factors,

d. An analysis of the potential effect of sudden temperature transients on waterlogged elements or elements with high internal gas pressure,

e. An analysis of temperature effects during anticipated operational transients that may cause bowing or other damage to fuel, control rods, or structure,

f. An analysis of the energy release and potential for a chemical reaction should physical burnout of fuel elements occur,*

g. An analysis of the energy release and resulting pressure pulse should waterlogged elements rupture and spill fuel into the coolant,*

h. An analysis of the behavior of fuel rods in the event coolant flow blockage is predicted.*

4. Spacer Grid and Channel Boxes

a. Dimensional stability considering thermal, chemical, and irradiation effects,

b. Spring loads for grids.

5. Fuel Assembly

a. Loads applied by core restraint system,

b. Analysis of combined shock (including LOCA) and seismic loading,

c. Loads applied in fuel handling, including misaligned handling tools.

6. Reactivity Control Assembly and Burnable Poison Rods

a. Internal pressure and cladding stresses during normal, transient, and accident conditions,

b. Thermal stability of the absorber material, including phase changes and thermal expansion,

c. Irradiation stability of the absorber material, taking into consideration gas release and swelling,

d. Potential for chemical interaction, including possible waterlogging rupture.

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*If this information is included in Chapter 15, it may be incorporated in this section by reference.

4.2.4 Testing and Inspection Plan

The testing and inspections to be performed to verify the design characteristics of the fuel system components, including clad integrity, dimensions, fuel enrichment, burnable poison concentration, absorber composition, and characteristics of the fuel, absorber, and poison pellets, should be described. Descriptions of radiographic inspections, destructive tests, fuel assembly dimensional checks, and the program for inspection of new fuel assemblies and new control rods to ensure mechanical integrity after shipment should be included. Where testing and inspection programs are essentially the same as for previously accepted plants, a referenced statement to that effect with an identification of the fabricator and a summary table of the important design and performance characteristics should be provided.

4.3 Nuclear Design

4.3.1 Design Bases

The design bases for the nuclear design of the fuel and reactivity control systems should be provided and discussed, including nuclear and reactivity control limits such as excess reactivity, fuel burnup, negative reactivity feedback, core design lifetime, fuel replacement program, reactivity coefficients, stability criteria, maximum controlled reactivity insertion rates, control of power distribution, shutdown margins, stuck rod criteria, rod speeds, chemical and mechanical shim control, burnable poison requirements, and backup and emergency shutdown provisions.

4.3.2 Description

A description of the nuclear characteristics of the design should be provided and should include the information indicated in the following sections.

4.3.2.1 Nuclear Design Description. Features of the nuclear design not discussed in specific subsections should be listed, described, or illustrated for appropriate times in the fuel cycle. These should include such areas as fuel enrichment distributions, burnable poison distributions, other physical features of the lattice or assemblies relevant to nuclear design parameters, delayed neutron fraction and neutron lifetimes, core lifetime and burnup, plutonium buildup, soluble poison insertion rates, and the relationship to cooldown or xenon burnout or other transient requirements.

4.3.2.2 Power Distribution. Full quantitative information on calculated "normal" power distributions, including distributions within typical assemblies, axial distributions, gross radial distributions (XY assembly patterns), and nonseparable aspects of radial and axial distributions should be presented.

143 050

A full range of both representative and limiting power density patterns related to representative and limiting conditions of such relevant parameters as power, flow, flow distribution, rod patterns, time in cycle (burnup and possible burnup distributions), cycle, burnable poison, and xenon should be covered in sufficient detail to ensure that normally anticipated distributions are fully described and that the effects of all parameters important in affecting distributions are displayed. This should include details of transient power shapes and magnitudes accompanying normal transients such as load following, xenon buildup, decay or redistribution, and xenon oscillation control. Describe the radial power distribution within a fuel pin and its variation with burnup if use is made of this in thermal calculations.

Discuss and assign specific magnitudes to errors or uncertainties that may be associated with these calculated distributions and present the experimental data, including results from both critical experiments and operating reactors that back up the analysis, likely distribution limits, and assigned uncertainty magnitudes. Experimental checks to be made on this reactor and the criteria for satisfactory results should be discussed.

The design power distributions (shapes and magnitudes) and the design peaking factors to be used in steady-state limit statements and transient analysis initial conditions should be given in detail. Include all relevant components and such variables as maximum allowable peaking factors vs. axial position or changes over the fuel cycle. Justify the selections by a discussion of the relationship of these design assumptions to the previously presented expected and limiting distributions and uncertainty analysis.

Describe the relationship of these distributions to the monitoring instrumentation, discussing in detail the adequacy of the number of instruments and their spatial deployment (including allowed failures); required correlations between readings and peaking factors, calibrations and errors, operational procedures and specific operational limits; axial and azimuthal asymmetry limits; limits for alarms, rod blocks, scrams, etc., to demonstrate that sufficient information is available to determine, monitor, and limit distributions associated with normal operation to within proper limits. Describe in detail all calculations, computer codes, and computers used in the course of operations that are involved in translating power-distribution-related measurements into calculated power distribution information. Give the frequency with which the calculations are normally performed and execution times of the calculations. Describe the input data required for the codes. Present a full quantitative analysis of the uncertainties associated with the sources and processing of information used to produce operational power distribution results. This should include consideration of allowed instrumentation failures.

4.3.2.3 Reactivity Coefficients. Full quantitative information on calculated reactivity coefficients, including fuel Doppler coefficient, moderator coefficients (density, temperature, pressure, void), and power

143 051

coefficient should be presented. The precise definitions or assumptions relating to parameters involved, e.g., effective fuel temperature for Doppler, distinction between intra- and interassembly moderator coefficients, parameters held constant in power coefficient, spatial variation of parameter, and flux weighting used, should be stated. The information should be primarily in the form of curves covering the full applicable range of the parameters (density, temperature, pressure, void, power) from cold startup through limiting values used in accident analyses. Quantitative discussions of both spatially uniform parameter changes and these nonuniform parameter and flux weighting changes appropriate to operational and accident analyses and the methods used to treat nonuniform changes in transient analysis should be included.

Sufficient information should be presented to illustrate the normal and limiting values of parameters appropriate to operational and accident states, considering cycle, time in cycle, control rod insertions, boron content, burnable poisons, power distribution, moderator density, etc. Potential uncertainties in the results of the calculations and experimental results that back up the analysis and assigned uncertainty magnitudes and experimental checks to be made in this reactor should be discussed. Where limits on coefficients are especially important, e.g., positive moderator coefficients in the power range, experimental checks on these limits should be fully detailed.

Present the coefficients actually used in transient analyses and show by reference to the previously discussed information and uncertainty analysis that suitably conservative values are used (1) for both beginning of life (BOL) and end of life (EOL) analyses, (2) where most negative or most positive (or least negative) coefficients are appropriate, and (3) where spatially nonuniform changes are involved.

<u>4.3.2.4 Control Requirements</u>. Tables and discussions relating to core reactivity balances for BOL, EOL, and, where appropriate, intermediate conditions should be provided. This should include consideration of such reactivity influences as control bank requirements and expected and minimum worths, burnable poison worths, soluble boron amounts and unit worths for various operating states, "stuck rod" allowance, moderator and fuel temperature and void defects, burnup and fission products, xenon and samarium poisoning, pH effects, permitted rod insertions at power and error allowances. Required and expected shutdown margin as a function of time in cycle, along with uncertainties in the shutdown margin and experimental confirmations from operating reactors should be presented and discussed.

Methods, paths, and limits for normal operational control involving such areas as soluble poison concentration and changes, control rod motion, power shaping rod (e.g., part length rod) motion, and flow change should be described fully. This should include consideration of cold, hot, and peak xenon startup, load following and xenon reactivity control, power shaping (e.g., xenon redistribution or oscillation control), and burnup.

143 052

4.3.2.5 Control Rod Patterns and Reactivity Worths. Full information on control rod patterns expected to be used throughout a fuel cycle should be presented. This should include details on separation into groups or banks if applicable, order and extent of withdrawal of individual rods or banks, limits, with justification, to be imposed on rod or bank positions as a function of power level and/or time in cycle or for any other reason, expected positions of rods or banks for cold critical, hot standby critical, and for full power for both BOL and EOL. Describe allowable deviations from these patterns for misaligned or stuck rods or for any other reason such as special power shaping. For the allowable patterns, including allowable deviations, indicate for various power and EOL and BOL conditions, the maximum worth of rods that might be postulated to be removed from the core in an ejection or drop accident and rods or rod banks that could be removed in rod withdrawal accidents, and give the worths of these rods as a function of position. Describe any experimental confirmation of these worths. Present maximum reactivity increase rates associated with these withdrawals. Describe fully and give the methods for calculating the scram reactivity as a function of time after scram signal, including consideration for Technical Specification scram times, stuck rods, power level and shape, time in cycle, and any other parameter important for bank reactivity worth and axial reactivity shape functions. For BWRs, provide criteria for control rod velocity limiters and control rod worth minimizers.

4.3.2.6 Criticality of Reactor During Refueling. The maximum value of k_{eff} for the reactor during refueling should be stated. Describe the basis for assuming that this maximum value will not be exceeded.

4.3.2.7 Stability. Information defining the degree of predicted stability with regard to xenon oscillations in both the axial direction and in the horizontal plane should be provided. If any form of xenon instability is predicted, include evaluations of higher mode oscillations. Indicate in detail the analytic and experimental bases for the predictions. Include an assessment of potential error in the predictions. Also, show how unexpected oscillations would be detectable before safety limits are exceeded.

Unambiguous positions regarding stability or lack thereof should be provided. That is, where stability is claimed, provide corroborating data from sufficiently similar power plants or provide commitments to demonstrate stability. Indicate criteria for determining whether the reactor will be stable or not. Where instability or marginal stability is predicted, provide details of how oscillations will be detected and controlled and provisions for protection against exceeding safety limits.

Analyses of the overall reactor stability against power oscillations (other than xenon) should be provided.

4.3.2.8 Vessel Irradiation. The neutron flux distribution and spectrum in the core, at core boundaries, and at the pressure vessel wall for appropriate times in the reactor life for NVT determinations should be provided.

4.3.3 Analytical Methods

A detailed description of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, and burnup effects should be provided. Computer codes used should be described in detail as to the name and the type of code, how it is used, and its validity based on critical experiments or confirmed predictions of operating plants. Code descriptions should include methods of obtaining parameters such as cross sections. Estimates of the accuracy of the analytical methods should be included.

4.3.4 Changes

Any changes in reactor core design features, calculational methods, data, or information relevant to determining important nuclear design parameters that depart from prior practice of the reactor designs should be listed along with affected parameters. Details of the nature and effects of the changes should be treated in appropriate subsections.

4.4 Thermal and Hydraulic Design

4.4.1 Design Bases

The design bases for the thermal and hydraulic design of the reactor should be provided, including such items as maximum fuel and clad temperatures and cladding-to-fuel gap characteristics as a function of burnup (at rated power, at design overpower, and during transients), critical heat flux ratio (at rated power, at design overpower, and during transients), flow velocities and distribution control, coolant and moderator voids, hydraulic stability, transient limits, fuel cladding integrity criteria, and fuel assembly integrity criteria.

4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

A description of the thermal and hydraulic characteristics of the reactor design should be provided and should include information indicated in the following sections.

4.4.2.1 Summary Comparison. A summary comparison of the thermal and hydraulic design parameters of the reactor with previously approved reactors of similar design should be provided. This should include, for example, primary coolant temperatures, fuel temperatures, maximum and average linear heat generation rates, critical heat flux ratios, critical heat flux correlations used, coolant velocities, surface heat fluxes, power densities, specific powers, surface areas, and flow areas.

4.4.2.2 Critical Heat Flux Ratios. The critical heat flux ratios for the core hot spot at normal full power and at design overpower conditions should be provided. State the critical heat flux correlation used, analysis techniques, method of use, method of employing peaking factors, and comparison with other correlations.

143 054

4.4.2.3 Linear Heat Generation Rate. The core-average linear heat generation rate (LHGR) and the maximum LHGR anywhere in the core should be provided. The method of utilizing hot channel factors and power distribution information to determine the maximum LHGR should be indicated.

4.4.2.4 Void Fraction Distribution. Curves showing the predicted radial and axial distribution of steam quality and steam void fraction in the core should be provided. State the predicted core average void fraction and the maximum void fraction anywhere in the core.

4.4.2.5 Core Coolant Flow Distribution. Coolant flow distribution and orificing and the basis on which orificing is designed relative to shifts in power production during core life should be described and discussed.

4.4.2.6 Core Pressure Drops and Hydraulic Loads. Core pressure drops and hydraulic loads during normal and accident conditions that are not addressed in Chapter 15 should be provided.

4.4.2.7 Correlation and Physical Data. The correlations and physical data employed in determining important characteristics such as heat transfer coefficients and pressure drop should be discussed.

4.4.2.8 Thermal Effects of Operational Transients. The capability of the core to withstand the thermal effects resulting from anticipated operational transients should be evaluated.

4.4.2.9 Uncertainties in Estimates. The uncertainties associated with estimating the peak or limiting conditions for thermal and hydraulic analysis (e.g., fuel temperature, clad temperature, pressure drops, and orificing effects) should be discussed.

4.4.2.10 Flux Tilt Considerations. Discuss the margin provided in the peaking factor to account for flux tilts to ensure that flux limits are not exceeded during operation. Describe plans for power reduction in the event of flux tilts and provide criteria for selection of a safe operating power level.

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

The thermal and hydraulic design of the reactor coolant system should be described in this section. The information indicated in the following sections should be included.

4.4.3.1 Plant Configuration Data. The following information on plant configuration and operation should be provided:

1. A description of the reactor coolant system, including isometric drawings that show the configuration and approximate dimensions of the reactor coolant system piping,

143 055

2. A listing of all valves and pipe fittings (elbows, tees, etc.) in the reactor coolant system,

3. Total coolant flow through each flowpath (total loop flow, core flow, bypass flow, etc.),

4. Total volume of each plant component, including ECCS compo.ents with sufficient detail in reactor vessel and the steam generator (for PWRs) to define each part (downcomer, lower plenum, upper head, etc.),

5. The flowpath length through each volume,

6. The height and liquid level of each volume,

7. The elevation of the bottom of each volume with respect to some reference elevation, preferably the centerline of the outer piping,

8. The line lengths and sizes of all safety injection lines,

9. Minimum flow areas of each component,

10. Steady-state pressure and tempera..re distribution throughout the system.

4.4.3.2 Operating Restrictions on Pumps. The operating restrictions that will be imposed on the coolant pumps to meet net positive suction head requirements should be stated.

4.4.3.3 Power-Flow Operating Map (BWR). For boiling water reactors, a power-flow operating map indicating the limits of reactor coolant system operation should be provided. This map should indicate the permissible operating range as bounded by minimum flow, design flow, maximum pump speed, and natural circulation.

4.4.3.4 Temperature-Power Operating Map (PWR). For pressurized water reactors, a temperature-power operating map should be provided. The effects of reduced core flow due to inoperative pumps, including system capability during natural circulation conditions, should be indicated.

4.4.3.5 Load-Following Characteristics. The load-following characteristics of the reactor coolant system and the techniques employed to provide this capability should be described.

4.4.3.6 Thermal and Hydraulic Characteristics Summary Table. A table summarizing the thermal and hydraulic characteristics of the reactor coolant system should be provided.

143 056

4.4.4 Evaluation

An evaluation of the thermal and hydraulic design of the reactor and the reactor coolant system should be provided. It should include the information indicated in the following sections.

4.4.4.1 Critical Heat Flux. The critical heat flux, departure from nucleate boiling, or critical power ratio correlation utilized in the core thermal and hydraulic analysis should be identified. The experimental basis for the correlation should be described, preferably by reference to documents available to the NRC. The applicability of the correlation to the proposed design should be discussed in the SAR. Particular emphasis should be placed on the effect of the grid spacer design, the calculational technique used to determine coolant mixing, and the effect of axial power distribution.

4.4.4.2 Core Hydraulics. The core hydraulics evaluation should include (1) a discussion of the results of flow model tests (with respect to pressure drop for the various flowpaths through the reactor and flow distributions at the core inlet), (2) the empirical correlation selected for use in analyses for both single-phase and two-phase flow conditions and the applicability over the range of anticipated reactor conditions, and (3) the effect of partial or total isolation of a loop.

4.4.4.3 Influence of Power Distribution. The influence of axial and radial power distributions on the thermal and hydraulic design should be discussed. An analysis to determine which fuel rods control the thermal limits of the reactor should be included.

4.4.4.4 Core Thermal Response. The thermal response of the core should be evaluated at rated power, at design overpower, and for expected transient conditions.

4.4.4.5 Analytical Methods. The analytical methods and data used to determine the reactor coolant system flow rate should be described. This should include classical fluid mechanics relationships and empirical correlations. The description should include both single-phase and two-phase fluid flow, as applicable. Estimates of the uncertainties in the calculations and the resultant uncertainty in reactor coolant system flow rate should be provided.

A comprehensive discussion of the analytical techniques used in evaluating the core thermal-hydraulics, including estimates of uncertainties, should be provided. This discussion should include such items as hydraulic instability, the application of hot spot factors and hot channel factors, subchannel hydraulic analysis, effects of crud (in the core and in the reactor coolant system), and operation with one or more loops isolated. Descriptions of computer codes may be included by reference to documents available to the NRC.

4.4.5 Testing and Verification

The testing and verification techniques to be used to ensure that the planned thermal and hydraulic design characteristics of the core and the reactor coolant system have been provided and will remain within required limits throughout core lifetime should be discussed. This discussion should address the applicable portions of Regulatory Guide 1.68, "Initial Test Program for Water-Cooled Reactor Power Plants." References to the appropriate portions of Chapter 14 are acceptable.

4.4.6 Instrumentation Requirements

The functional requirements for the instrumentation to be employed in monitoring and measuring those thermal-hydraulic parameters important to safety should be discussed. The requirements for in-core instrumentation to confirm predicted power density distribution and moderator temperature distributions, for example, should be included. Details of the instrumentation design and logic should be discussed in Chapter 7 of the SAR.

The vibration and loose-parts monitoring equipment to be provided in the plant should be described. The procedures to be used to detect excessive vibration and the occurrence of loose parts should be discussed.

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

For the purpose of this section, the control rod drive system includes the control rod drive mechanism (CRDM) and extends to the coupling interface with the reactivity control (poison) elements in the reactor vessel. It does not include the electrical and hydraulic systems necessary for actuating the CRDMs. The information described below should be provided.

1. Materials Specifications

a. Provide a list of the materials and their specifications for each component of the control rod drive mechanism. Furnish information regarding the mechanical properties of any material not included in Appendix I to Section III of the ASME B&PV Code or Regulatory Guide 1.85, "Materials Code Case Acceptability - ASME Section III Division 1," and provide justification for the use of such material.

b. State whether any of the following materials that have a yield strength greater than 90,000 psi are being used: cold-worked austenitic stainless steels, precipitation hardenable stainless steels, or hardenable martensitic stainless steels. If such materials are employed, identify their usage and provide evidence that stress-corrosion cracking will not occur during service life in components fabricated from the materials.

143 058

2. Austenitic Stainless Steel Components

a. Provide a description of the processes, inspections, and tests on austenitic stainless steel components to ensure freedom from increased susceptibility to intergranular stress-corrosion cracking caused by sensitization. If special processing or fabrication methods subject the materials to temperatures between 800 and 1500°F, or involve slow cooling from temperatures over 1500°F, describe the processing or fabrication methods and provide justification to show that such treatment will not cause susceptibility to intergranular stress-corrosion cracking. Indicate the degree of conformance to the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Provide justification for any deviations from these recommendations.

b. State the procedures and requirements that will be applied to prevent hot cracking in austenitic stainless steel welds, especially those procedures and requirements to control the delta ferrite content in weld filler metal and in completed welds. Indicate the degree of conformance to the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." Provide justification for any deviations from these recommendations.

3. Other Materials

The tempering temperature of hardenable martensitic stainless steels and the aging temperature and aging time of precipitation-hardening stainless steels should be described. The processing and treatment of other special purpose materials such as cobalt-base alloys (Stellites), Inconels, Colmonoys, and Graphitars should be described.

4. Cleaning and Cleanliness Control

Provide details of the steps that will be taken in protecting austenitic stainless steel materials and parts of these systems during fabrication, shipping, and onsite storage to ensure that all cleaning solutions, processing compounds, degreasing agents, and detrimental contaminants are completely removed and that all parts are dried and properly protected following any flushing treatment with water. Indicate the degree of conformance to the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." Provide justification for any deviations from these recommendations.

4.5.2 Reactor Internals Materials

This section should discuss the materials used for reactor internals and should include the information described below.

4.5.2.1 Materials Specifications. Provide a list of the materials and their specifications for major components of the reactor internals. Include materials treated to enhance corrosion resistance, strength, and hardness. Furnish information regarding the mechanical properties of any material not included in Appendix I to Section III of the ASME B&PV Code and provide justification for the use of such material.

4.5.2.2 Controls on Welding. Indicate the controls that will be used when welding reactor internals components, and provide assurance that such welds will meet the acceptance criteria of Article NG-5000 of ASME B&PV Code Section III or alternative acceptance criteria that provide an acceptable level of safety.

4.5.2.3 Nondestructive Examination of Tubular Products and Fittings. Indicate that the nondestructive examination procedures used for the examination of tubular products conform to the requirements of the ASME B&PV Code. Provide justification for any deviations from these requirements.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel Components. Indicate the degree of conformance with the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel;" Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal;" and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." If alternative measures are used, show that they will provide the same assurance of component integrity as would be achieved by following the recommendations of the guides. Indicate the maximum yield strength of all cold worked stainless steels used in the reactor internals.

4.5.2.5 Other Materials. The tempering temperature of hardenable martensitic stainless steels and the aging temperatures and aging time of precipitation-hardening stainless steels should be described. The processing and treatment of other special purpose materials such as cobalt-base alloys (Stellites), Inconels, and Colmonoys should be described.

4.6 Functional Design of Reactivity Control Systems

Information should be presented to establish that the control rod drive system (CRDS), which includes the essential ancillary equipment and hydraulic systems, is designed and installed to provide the required functional performance and is properly isolated from other equipment. Additionally, information should be presented to establish the bases for assessing the combined functional performance of all the reactivity control systems to mitigate the consequences of anticipated transients and postulated accidents.

These reactivity control systems include, in addition to the CRDS and the emergency core cooling system (ECCS), the chemical and volume control system (CVCS) and the emergency boration system (EBS) for pressurized water reactors and the standby liquid control system (SLCS) and the recirculation flow control system (RFCS) for boiling water reactors.

143 060

4.6.1 Information for CRDS

Information submitted should include drawings of the rod drive mechanism, layout drawings of the collective rod drive system, process flow diagrams, piping and instrumentation diagrams, component descriptions and characteristics, and a description of the functions of all related ancillary equipment and hydraulic systems. This information may be presented in conjunction with the information requested for Section 3.9.4.

4.6.2 Evaluations of the CRDS

Failure mode and effects analyses of the CRDS should be presented in tabular form with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the CRDS, which for purposes of these evaluations includes all essential ancillary equipment and hydraulic systems, can perform the intended safety functions with the loss of any single active component.

These evaluations and assessments should establish that all essential elements of the CRDS are identified and provisions made for isolation from nonessential CRDS elements. It should be established that all essential equipment is amply protected from common mode failures such as failure of moderate- and high-energy lines.

4.6.3 Testing and Verification of the CRDS

A functional testing program should be presented. This should include rod insertion and withdrawal tests, thermal and fluid dynamic tests simulating postulated operating and accident conditions, and test verification of the CRDS with imposed single failures, as appropriate.

Preoperational and initial startup test programs should be presented. The objectives, test methods, and acceptance criteria should be included.

4.6.4 Information for Combined Performance of Reactivity Systems

Information consisting of piping and instrumentation diagrams, layout drawings, process diagrams, failure analyses, descriptive material, and performance evaluations related to specific evaluations of the CVCS, the SLCS, and the RFCS is presented in other sections of the safety analysis report, e.g., 9.3.4 and 9.3.5. This section should include sufficient plan and elevation layout drawings to provide bases for establishing that the reactivity control systems (CRDS, ECCS, CVCS, SLCS, RFCS, EBS) when used in single or multiple redundant modes are not vulnerable to common mode failures.

Evaluations pertaining to the response of the plant to postulated process disturbances and to postulated malfunctions or failures of equipment are presented in Chapter 15, "Accident Analyses." This section should include a list of all the postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems for preventing or mitigating each accident. The related reactivity systems should also be tabulated.

4.6.5 Evaluations of Combined Performance

Evaluations of the combined functional performance for accidents where two or more reactivity systems are used should be presented. The neutronic, fluid dynamic, instrumentation, controls, time sequencing, and other process-parameter-related features are presented primarily in Chapters 4, 7, and 15 of the safety analysis report. This section should include failure analyses to demonstrate that the reactivity control systems used redundantly are not susceptible to common mode failures. These failure analyses should consider failures originating within each reactivity control system and from plant equipment other than reactivity systems and should be presented in tabular form with supporting discussion and logic.

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter of the SAR should provide information regarding the reactor coolant system and systems connected to it. Special consideration should be given to the reactor coolant system and pressure-containing appendages out to and including isolation valving which is the "reactor coolant pressure boundary" (RCPB), as defined in paragraph 50.2(v) of 10 CFR Part 50.

Evaluations, together with the necessary supporting material, should be submitted to show that the reactor coolant system is adequate to accomplish its intended objective and to maintain its integrity under conditions imposed by all foreseeable reactor behavior, either normal or accident conditions. The information should permit a determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete and all the evaluations needed have been made. Evaluations included in other chapters that have a bearing on the reactor coolant system should be referenced.

5.1 Summary Description

A summary description of the reactor coolant system and its various components should be provided. The description should indicate the independent and interrelated performance and safety functions of each component. Include a tabulation of important design and performance characteristics.

5.1.1 Schematic Flow Diagram

A schematic flow diagram of the reactor coolant system denoting all major components, principal pressures, temperatures, flow rates, and coolant volume under normal steady-state full power operating conditions should be provided.

5.1.2 Piping and Instrumentation Diagram

Provide a piping and instrumentation diagram of the reactor coolant system and connected systems delineating the following:

1. The extent of the systems located within the containment,

2. The points of separation between the reactor coolant (heat transport) system and the secondary (heat utilization or removal) system, and

3. The extent of isolability of any fluid system as provided by the use of isolation valves between the radioactive and nonradioactive sections of the system, isolation valves between the RCPB and connected systems, and passive barriers between the RCPB and other systems.

143 063

5.1.3 Elevation Drawing

Provide an elevation drawing showing principal dimensions of the reactor coolant system in relation to the supporting or surrounding concrete structures from which a measure of the protection afforded by the arrangement and the safety considerations incorporated in the layout can be gained.

5.2 Integrity of Reactor Coolant Pressure Boundary

This section should present discussions of the measures to be employed to provide and maintain the integrity of the reactor coolant pressure boundary (RCPB) for the plant design lifetime.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR §50.55a. A table showing compliance with the regulations of 10 CFR §50.55a, "Codes and Standards," should be provided. This table should identify pressure vessel components, piping, pumps, and valves. The applicable component code, code edition, code addenda, and, when required, the component order date of each ASME Section III, Class 1 component within the reactor coolant pressure boundary may be identified by reference to the table of structures, systems, and components in Section 3.2 of the SAR; alternatively, they may be included in this section of the SAR. In the event there are cases wherein conformance to the regulations of §50.55a would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, a complete description of the circumstances resulting in such cases and the basis for proposed alternative requirements should be provided. Describe how an acceptable level of safety and quality will be provided by the proposed alternative requirements.

5.2.1.2 Applicable Code Cases. Provide a list of ASME Code Case interpretations that will be applied to components within the reactor coolant pressure boundary. Each component to which a Code Case has been applied should be identified by Code Case number, revision, and title. Caution is advised in the use of Code Cases to ensure that the applicable revision of a Code Case is identified for each component application. Regulatory Guides 1.84, "Design and Fabrication Code Case Acceptability - ASME Section III Division 1," and 1.85, "Materials Code Case Acceptability - ASME Section III Division 1," list those Section III, Division 1, ASME Code Cases that are generally acceptable. The section should indicate the extent of conformance with the recommendations of Regulatory Guides 1.84 and 1.85. If Code Cases other than those listed are used, show that their use will result in as acceptable a level of quality and safety for the component as would be achieved by following the recommendation of the guides.

143 064

5.2.2 Overpressurization Protection

The information cited below should be provided to accommodate an evaluation of the systems that protect the RCPB and the secondary side of steam generators from overpressurization. These systems include all pressure-relieving devices (safety and relief valves) for:

1. The reactor coolant system,

2. The primary side of auxiliary or emergency systems connected to the reactor coolant system,

3. Any blowdown or heat dissipation systems connected to the discharge of these pressure-relieving devices, and

4. The secondary side of steam generators.

5.2.2.1 Design Bases. Provide the design bases on which the functional design of the overpressure protection system was established. Identify the postulated events or transients on which the design requirements are based, including:

1. The extent of simultaneous occurrences,

2. The assumptions regarding initial plant conditions and system parameters, and

3. A list of all systems that could initiate during the postulated event and the initiating and trip signals.

5.2.2.2 Design Evaluation. An evaluation of the functional design of the overpressurization system should be provided. Present an analysis of the capability of the system to perform its function. Describe the analytical model used in the analysis and discuss the bases for its validity. Discuss and justify the assumptions used in the analysis, including the plant initial conditions and system parameters. List the systems and equipment assumed to operate and describe their performance characteristics. Provide studies that show the sensitivity of the performance of the system to variations in these conditions, parameters, and performance.

5.2.2.3 Piping and Instrumentation Diagrams. Provide piping and instrumentation diagrams for the overpressure protection system showing the number and location of all components, including valves, piping, tanks, instrumentation, and controls. Connections and other interfaces with other systems should be indicated.

5.2.2.4 Equipment and Component Description. Describe the equipment and components of the overpressure protection system, including schematic drawings of the safety and relief valves and a discussion of how the valves operate. Identify the significant design parameters for each component, including the design, throat area, capacity, and set point of the valves and the diameter, length, and routing of piping. List the design parameters (e.g., pressure and temperature) for each component. Specify the number and type of operating cycles for which each component is designed. The environmental conditions (e.g., temperature and humidity) for which the components are designed should also be specified.

5.2.2.5 Mounting of Pressure-Relief Devices. Describe the design and installation details of the mounting of the pressure-relief devices within the reactor coolant pressure boundary and the secondary side of steam generators. Specify the design bases for the assumed loads (i.e., thrust, bending, and torsion) imposed on the valves, nozzles, and connected piping in the event all valves discharge. Describe how these loads can be accommodated; include a listing of these loads and resulting stresses. Material contained in Section 3.9.3.3 may be incorporated by reference.

5.2.2.6 Applicable Codes and Classification. Identify the applicable industry codes and classifications applied to the system.

5.2.2.7 Material Specification. The material specifications for each component should be identified.

5.2.2.8 Process Instrumentation. Identify all process instrumentation.

5.2.2.9 System Reliability. The reliability of the system and the consequences of failures should be discussed.

5.2.2.10 Testing and Inspection. Identify the tests and inspections to be performed (1) prior to operation and during startup which demonstrate the functional performance and (2) as inservice surveillance to ensure continued reliability.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Material Specifications. Provide a list of specifications for the principal ferritic materials, austenitic stainless steels, and nonferrous metals, including bolting and weld materials, to be used in fabricating and assembling each component (e.g., vessels, piping, pumps, and valves) that is part of the reactor coolant pressure boundary (RCPB), excluding the reactor pressure vessel. Identify the grade or type and final metallurgical condition of the material placed in service.

5.2.3.2 Compatibility with Reactor Coolant. Provide the following information relative to compatibility of the reactor coolant with the materials of construction and the external insulation of the RCPB:

1. PWR reactor coolant chemistry (for PWRs only). Provide a description of the chemistry of the reactor coolant and the additives (such as inhibitors). Describe water chemistry, including maximum allowable content

of chloride, fluoride, and oxygen and permissible content of hydrogen and soluble poisons. Discuss methods to control water chemistry, including pH.

2. BWR reactor coolant chemistry (for BWRs only). Describe the chemistry of the reactor coolant and the methods for maintaining coolant chemistry. Provide sufficient information about allowable range and maximum allowable chloride and fluoride contents, maximum allowable conductivity, pH range, location of conductivity meters, performance monitoring, and other details of the coolant chemistry program to indicate whether coolant chemistry will be maintained at a level comparable to the recommendations in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."

3. Compatibility of construction materials with reactor coolant. Provide a list of the materials of construction exposed to the reactor coolant and a description of material compatibility with the coolant, contaminants, and radiolytic products to which the materials may be exposed. If nonmetallics are exposed to the reactor coolant, include a description of the compatibility of these materials with the coolant.

4. Compatibility of construction materials with external insulation and reactor coolant. Provide a list of the materials of construction of the RCPB and a description of their compatibility with the external insulation, especially in the event of a coolant leakage. Provide sufficient information about the selection, procurement, testing, storage, and installation of any nonmetallic thermal insulation for austenitic stainless steel to indicate whether the concentrations of chloride, fluoride, sodium, and silicate in thermal insulation will be within the ranges recommended in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Provide information on the leachable contaminants in insulation on nonaustenitic piping.

5.2.3.3 Fabrication and Processing of Ferritic Materials. Provide the following information relative to fabrication and processing of ferritic materials used for components of the RCPB:

1. Fracture toughness. In regard to fracture toughness of the ferritic materials, including bolting materials for components (e.g., vessels, piping, pumps, and valves) of the RCPB, indicate how compliance with the test and acceptance requirements of Appendix G to 10 CFR Part 50 and with Section NB-2300 and Appendix G of the ASME Code, Section III, is achieved. Submit the fracture toughness data in tabular form, including information regarding the calibration of instruments and equipment (FSAR).

2. Control of welding. Provide the following information relative to control of welding of ferritic materials used for components of the RCPB:

a. Sufficient information regarding the avoidance of cold cracking during welding of low-alloy steel components of the RCPB to

143 067

indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guides 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," and 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Provide details on proposed minimum preheat temperature and maximum interpass temperature during procedure qualification and production welding. Provide information on the moisture control for low-hydrogen, covered-arc-welding electrodes.

b. Sufficient information for electroslag welds in the low-alloy steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.34, "Control of Electroslag Weld Properties." Provide details on the control of welding variables and the metallurgical tests required during procedure qualification and production welding.

c. In regard to welding and weld repair during fabrication and assembly of ferritic steel components of the RCPB, provide sufficient details for welder qualification for areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility."

3. Nondestructive examination. Provide sufficient information on nondestructive examination of ferritic steel tubular products (pipe, tubing, flanges, and fittings) for components of the RCPB to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) will be in conformance with the requirements of the ASME Code.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels. Provide the following information relative to fabrication and processing of austenitic stainless steels for components of the RCPB:

1. Avoidance of stress-corrosion cracking. Provide the following information relative to avoidance of stress-corrosion cracking of austenitic stainless steels for components of the RCPB during all stages of component manufacture and reactor construction:

a. Sufficient details about the avoidance of sensitization during fabrication and assembly of austenitic stainless steel components of the RCPB to indicate whether the degree of freedom from sensitization will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." Provide a description of materials (including provision for 5% minimum delta ferrite when required), welding and heat treating processes, inspections, and tests.

143 068

b. Sufficient details about the process controls to minimize exposure to contaminants capable of causing stress-corrosion cracking of austenitic stainless steel components of the RCPB to show whether the process controls will provide, during all stages of component manufacture and reactor construction, a degree of surface cleanliness comparable to that obtainable by following the recommendations of Regulatory Guide 1.44 and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

c. Characteristics and mechanical properties of cold-worked austenitic stainless steels for components of the RCPB. If such steels are employed at yield strength levels greater than 90,000 psi, provide assurance that they will be compatible with the reactor coolant.

2. Control of welding. Provide the following information relative to the control of welding of austenitic stainless steels for components of the RCPB:

a. Sufficient information about the avoidance of hot cracking (fissuring) during weld fabrication and assembly of austenitic stainless steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." Describe the requirements regarding welding procedures and the amount of and method of determining delta ferrite in weld filler metals and in qualification welds.

b. Sufficient information about electroslag welds in austenitic stainless steel components of the RCPB to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.34. Provide details on the control of welding variables and the metallurgical tests required during procedure qualification and production welding.

c. In regard to welding and weld repair during fabrication and assembly of austenitic stainless steel components of the RCPB, provide sufficient details about welder qualification for areas of limited accessibility, requalification, and monitoring of production welding for adherence to welding qualification requirements to indicate whether the degree of weld integrity and quality will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.71.

3. Nondestructive examination. Provide sufficient information about the program for nondestructive examination of austenitic stainless steel tubular products (pipe, tubing, flanges, and fittings) for components of the RCPB to indicate whether detection of unacceptable defects (regardless of defect shape, orientation, or location in the product) will be in conformance with the requirement of the ASME Code.

5.2.4 Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

This section should discuss the inservice inspection and testing program for the NRC Quality Group A components (ASME Boiler and Pressure Vessel Code, Section III, Class 1 components). Provide sufficient detail to show that the inservice inspection program meets the requirements of Section XI of the ASME Code. Areas to be discussed should include:

1. System boundary subject to inspection, including associated component supports, structures, and bolting,

2. Arrangement of systems and components to provide accessibility,

3. Examination techniques and procedures, including any special techniques and procedures that might be used to meet the Code requirement,

4. Inspection intervals,

5. Inservice inspection program categories and requirements,

6. Evaluation of examination results, and

7. System leakage and hydrostatic pressure tests

In the FSAR, a detailed inservice inspection program, including information on areas subject to examination, method of examination, and extent and frequency of examination, should be provided in Chapter 16, "Technical Specifications."

5.2.5 Detection of Leakage Through Reactor Coolant Pressure Boundary

The program should be described and sufficient leak detection system information should be furnished to indicate the extent to which the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," have been followed.

Specifically, provide information that will permit comparison with the regulatory positions of the guide, giving a detailed description of the systems employed, their sensitivity and response time, and the reliance placed on their proper functioning. Also, the limiting leakage conditions that will be included in the technical specifications should be provided.

Identify the leakage detection systems that are designed to meet the sensitivity and response guidelines of Regulatory Guide 1.45. Describe these systems as discussed in Section 7.5, "Safety-Related Display Instrumentation." Also, identify those systems that are used for alarm as an indirect indication of leakage, and provide the design criteria.

Describe how signals from the various leakage detection systems are correlated to provide information to the plant operators on conditions of quantitative leakage flow rate.

Discuss the provisions for testing and calibration of the leak detection systems.

5.3 Reactor Vessels

5.3.1 Reactor Vessel Materials

This section should contain pertinent data in enough detail to provide assurance that the materials, fabrication methods, and inspection techniques used for the reactor vessel conform to all applicable regulations.

The PSAR should describe the specifications and criteria to be applied, whereas the FSAR should demonstrate that these requirements have been met.

5.3.1.1 Material Specifications. List all materials in the reactor vessel and its appurtenances and provide the applicable material specifications, making appropriate references to Section 5.2.3. If any materials other than those listed in Appendix I to the ASME Boiler and Pressure Vessel Code, Section III, are used, provide the data called for under Appendix IV for approval of the new material. Information provided in Section 5.2.3.1 may be incorporated by reference.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication. Describe the manufacture of the product forms and the methods used to fabricate the vessel. Discuss any special or unusual processes used, and show that they will not compromise the integrity of the reactor vessel.

5.3.1.3 Special Methods for Nondestructive Examination. Describe in detail all special procedures for detecting surface and internal discontinuities with emphasis on procedures that differ from those in Section III of the Code. Pay particular attention to calibration methods, instrumentation, method of application, sensitivity, reliability, reproducibility, and acceptance standards.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels. Making appropriate references to Section 5.2.3, describe controls on welding, composition, heat treatments, and similar processes covered by regulatory guides to verify that these recommendations or equivalent controls are employed. The following regulatory guides should be addressed:

Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal;"

Regulatory Guide 1.34, "Control of Electroslag Weld Properties;"

Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components;"

Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel;"

Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel;"

Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility;" and

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

5.3.1.5 Fracture Toughness. Describe the fracture testing and acceptance criteria specified for materials of the reactor vessel. In particular, describe how the toughness requirements of Appendix G to 10 CFR Part 50 will be met.

In the FSAR, report the results of fracture toughness tests on all ferritic materials of the reactor vessel, and demonstrate that the material toughness meets all requirements.

5.3.1.6 Material Surveillance. Describe the material surveillance program in detail. Provide assurance that the program meets the requirements of Appendix H to 10 CFR Part 50. In particular, consider the following subjects:

1. Basis for selection of material in the program,

2. Number and type of specimens in each capsule,

3. Number of capsules and proposed withdrawal schedule,

4. Neutron flux and fluence calculations for the vessel wall and surveillance specimens,

5. Expected effects of radiation on the vessel wall materials and the basis for this estimation, and

6. Location of capsules, method of attachment, and provisions to ensure that capsules will be retained in position throughout the lifetime of the vessel.

5.3.1.7 Reactor Vessel Fasteners. Describe the materials and design of fasteners for the reactor vessel closure. Include enough detail regarding materials property requirements, nondestructive evaluation procedures, lubricants or surface treatments, and protection provisions to show that the recommendations of Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," or equivalent measures, are followed.

143 072

In the FSAR, include the results of mechanical property and toughness tests to demonstrate that the material conforms to these recommendations or their equivalent.

5.3.2 Pressure-Temperature Limits

This section should describe the bases for setting operational limits on pressure and temperature for normal, upset, and test conditions. It should provide detailed assurance that Appendices G and H to 10 CFR Part 50 will be complied with throughout the life of the plant.

5.3.2.1 Limit Curves. Provide limits on pressure and temperature for the following conditions:

1. Preservice system hydrostatic tests,

2. Inservice leak and hydrostatic tests,

3. Normal operation, including heatup and cooldown, and

4. Reactor core operation.

If procedures or criteria other than those recommended in the ASME Boiler and Pressure Vessel Code are used, show that equivalent safety margins are provided.

In the PSAR, describe the bases used to determine these limits, and provide typical curves with temperatures relative to the RT_{NDT} (as defined in paragraph NB-2331 of Section III of the ASME Code) of the limiting material.

In the FSAR and technical specifications, include the actual material toughness test results, and provide limits based on these properties and the predicted effects of irradiation. Describe the bases used for the prediction, and indicate the extent to which the recommendations of Regulatory Guide 1.99 are followed.

Describe the procedures that will be used to update these limits during service, taking into account radiation effects.

5.3.2.2 Operating Procedures. Compare the pressure-temperature limits in Section 5.3.2.1 with intended normal operating procedures, and show that the limits will not be exceeded during any foreseeable upset condition.

5.3.3 Reactor Vessel Integrity

This section should contain any important information about vessel integrity not covered in other sections. In addition, it should summarize the major considerations in achieving reactor vessel safety and describe the factors contributing to the vessel's integrity.

The introductory material should identify the reactor vessel designer and manufacturer and should describe their experience.

5.3.3.1 Design. Include a brief description of the basic design, preferably with a simple schematic showing materials, construction features, and fabrication methods. Summarize applicable design codes and bases. Reference other sections of the SAR as appropriate.

5.3.3.2 Materials of Construction. Note briefly the materials used and describe any special requirements to improve their properties or quality. Emphasize the reasons for selection and provide assurance of suitability.

5.3.3.3 Fabrication Methods. Summarize the fabrication methods. Describe the service history of vessels constructed using these methods and the vessel supplier's experience with the procedures.

5.3.3.4 Inspection Requirements. Summarize the inspection requirements, paying particular attention to the level of initial integrity. Describe any examination methods used that are in addition to the minimum requirements of Section III of the ASME Code.

5.3.3.5 Shipment and Installation. Summarize the means used to protect the vessel so that its as-manufactured integrity will be maintained during shipment and installation. Reference other sections of the SAR as appropriate.

5.3.3.6 Operating Conditions. Summarize the operational limits that will be specified to ensure vessel safety. Provide a basis for concluding that vessel integrity will be maintained during the most severe postulated transients, or reference other appropriate SAR sections.

5.3.3.7 Inservice Surveillance. Making appropriate reference to Section 5.2.4, summarize the inservice inspection and material surveillance programs and explain why they are adequate.

5.4 Component and Subsystem Design

This section should present discussions of the performance requirements and design features to ensure overall safety of the various components within the reactor coolant system and subsystems closely allied with the reactor coolant system.

Because these components and subsystems differ for various types and designs of reactors, the Standard Format does not assign specific subsection numbers to each of these components or subsystems. The applicant should provide separate subsections (numbered 5.4.1 through 5.4.X) for each principal component or subsystem. The discussion in each subsection should present the design bases, description, evaluation, and necessary tests and inspections for the component or subsystem, including a discussion of the radiological considerations for each subsystem from a viewpoint of how radiation affects the operation of the subsystem and from a viewpoint of how radiation levels affect the operators and capabilities of operation and maintenance. Appropriate details of the mechanical design should be described in Sections 3.7, 3.9, and 5.2.

The following paragraphs provide examples of components and subsystems that should be discussed as appropriate to the individual plant and identify some specific information that should be provided in addition to the items identified above.

5.4.1 Reactor Coolant Pumps

In addition to the discussion of design bases, description, evaluations, and tests and inspections, the provisions taken to preclude rotor overspeeding of the reactor coolant pumps in the event of a design basis LOCA should be discussed.

5.4.1.1 Pump Flywheel Integrity (PWR). The applicant should provide explicit information to indicate the extent to which the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," are followed in the design, testing, and inservice inspection of the reactor coolant pump flywheels.

5.4.2 Steam Generators (PWR)

The information provided should include estimates of design limits for radioactivity levels in the secondary side of the steam generators during normal operation and the bases for these estimates. The potential effects of tube ruptures should be discussed.

Provide the steam generator design criteria used to prevent unacceptable tube damage from flow-induced vibration and cavitation. Information included in Section 3.9.3 should be referenced in this section. The following specific information should be included:

1. The design conditions and transients that will be specified in the design of the steam generator tubes and the operating condition category selected (e.g., upset, emergency, or faulted) that defines the allowable stress intensity limits to be used and the justification for this selection.

2. The extent of tube-wall thinning that could be tolerated without exceeding the allowable stress intensity limits defined above under the postulated condition of a design basis pipe break in the reactor coolant pressure boundary or a break in the secondary piping during reactor operation.

5.4.2.1 Steam Generator Materials. This section should contain information on the selection and fabrication of Code Class 1 and 2 steam generator materials (including those that are part of the reactor coolant pressure boundary), the design aspects of the steam generator that affect

143 075

materials performance, and the compatibility of the steam generator materials with the primary and secondary coolant.

1. Selection and Fabrication of Materials. Making appropriate references to Section 5.2.3, provide information on the selection and fabrication of materials for Code Class 1 and 2 components of the steam generators, including tubing, tube sheet, channel head casting or plate, tube sheet and channel head cladding, forged nozzles, shell pressure plates, access plates (manway and handhole), and bolting. Indicate the method used to fasten tubes to the tube sheet and show that it meets the requirements of Sections III and IX of the ASME Code. Include the extent of tube expansion and the methods of expansion used. Describe onsite cleaning and cleanliness control provisions, and show that they produce results equivalent to those obtained by following the recommendations of Regulatory Guide 1.37 and ANSI Standard N45.21-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants." For steam generators that are shipped partially assembled, include a discussion of the techniques used to maintain cleanliness during shipment and final assembly. List the Code Cases used in material selection. Technical justification for any Code Cases not listed in Regulatory Guide 1.85, "Materials Code Case Acceptability - ASME Section III Division 1," should be provided.

Provide information on the fracture toughness properties of ferritic materials, making appropriate references to Section 5.2.3. Sufficient information on materials for Class 1 components should be given to show that they meet the requirements of Article NB-2300 and Appendix G of Section III of the ASME Code. Sufficient information on Class 2 materials should be provided to show the extent to which they meet the requirements of Article NC-2300 of Section III of the Code.

2. Steam Generator Design. Provide information on those aspects of steam generator design that may affect the performance of steam generator materials. Describe the methods used to avoid extensive crevice areas where the tubes pass through the tube sheet and tubing supports.

3. Compatibility of the Steam Generator Tubing with the Primary and Secondary Coolant. Provide information on the compatibility of the steam generator tubing with both the primary and secondary coolant.

4. Cleanup of Secondary Side. Describe the procedures and methods used to remove surface deposits, sludge, and excessive corrosion products in the secondary side.

<u>5.4.2.2 Steam Generator Inservice Inspection</u>. In this section, the PSAR should describe the provisions in the design of the steam generators to permit inservice inspection of all Code Class 1 and 2 components, including individual steam generator tubes. The FSAR should describe detailed plans for baseline and inservice inspections of all Code Class 1 and 2 components making appropriate references to Section 5.2.4.

143 076

1. Compliance with Section XI of the ASME Code. Provide sufficient information on the proposed inservice inspection program for Code Class 1 and 2 components of the steam generators to show that it complies with the edition of Section XI of the ASME Code, Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," required by 10 CFR 50.55a, paragraph g.

2. Program for Inservice Inspection of Steam Generator Tubing. Provide sufficient information in the FSAR on the inservice inspection program for steam generator tubing to show that it will be at least as effective as the program recommended in Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." The information provided should include a description of the equipment, procedures, sensitivity of the examination, and recording methods; criteria used to select tubes for examination; inspection intervals; and actions that will be taken if defects are found (including criteria for plugging defective tubes).

5.4.3 Reactor Coolant Piping

The section on reactor coolant piping should present an overall description of this system, making appropriate references to detailed information on criteria, methods, and materials provided in Chapter 3 and Section 5.2.3. The discussion should include the provisions taken during design, fabrication, and operation to control those factors that contribute to stress corrosion cracking.

5.4.4 Main Steam Line Flow Restrictions

5.4.5 Main Steam Line Isolation System

Include discussion of provisions, such as seal systems, taken to reduce the potential leakage of radioactivity to the environment in the event of a main steam line break.

5.4.6 Reactor Core Isolation Cooling System

5.4.6.1 Design Bases. A summary description of the reactor core isolation cooling (RCIC) system should be provided. The RCIC system design bases and criteria for both the steamside and pumpside should be specified, and in particular discuss:

1. The design bases with respect to General Design Criteria 34, 55, 56, and 57.

2. Design bases concerned with reliability and operability requirements. The design bases for the manual operations required to operate the system should be described.

3. Design bases for RCIC operation following a loss of offsite power event.

4. The design bases established for the purpose of protecting the RCIC system from physical damage. This discussion should include the design bases for the RCIC system support structure and for protection against incidents that could fail RCIC and high pressure core spray (HPCS) jointly.

5.4.6.2 System Design. This section should include:

1. Schematic Piping and Instrumentation Diagrams. Provide a description of the RCIC system. Provide piping and instrumentation diagrams showing all components, piping, points where connecting systems and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation. Provide a complete description of component interlocks. Provide a diagram showing temperatures, pressures, and flow rates for RCIC operation.

2. Equipment and Component Descriptions. Describe each component of the system. Identify the significant design parameters for each component. State the design pressure and temperature of components for various portions of the system and explain the bases for their selection.

3. Applicable Codes and Classifications. Identify the applicable industry codes and classifications for the system design.

4. System Reliability Considerations. Discuss the provisions incorporated in the design to ensure that the system will operate when needed and will deliver the required flow rates.

5. Manual Actions. Discuss all manual actions required to be taken by an operator in order for the RCIC system to operate properly, assuming all components are operable. Identify any actions that are required to be taken from outside the control room. Repeat this discussion for the most limiting single failure in the combined RCIC and HPCS system.

5.4.6.3 Performance Evaluation. Provide an evaluation of the ability of the RCIC system to perform its function. Describe the analytical methods used and clearly state all assumptions.

5.4.6.4 Preoperational Testing. The proposed preoperational test program should be discussed. The discussion should identify test objectives, method of testing, and test acceptance criteria.

5.4.7 Residual Heat Removal System

5.4.7.1 Design Bases. A summary description of the residual heat removal (RHR) system should be provided. Nuclear plants employing the same RHR system design that are operating or have been licensed should be referenced. The design basis should be specified, including:

1. Functional design bases, including the time required to reduce the reactor coolant system (RCS) temperature to approximately 212°F, and

143 078

to a temperature that would permit refueling. The design basis times should be presented for the case where the entire RHR system is operable and for the case with the most limiting single failure in the RHR system.

2. The design bases for the isolation of the RHR system from the RCS. These isolation design bases should include any interlocks that are provided. The design bases regarding prevention of RHR pump damage in event of closure of the isolation valves should be discussed.

3. The design basis for the pressure relief capacity of the RHR system. These design bases should consider limiting transients, equipment malfunctions, and possible operator errors during plant startup and cooldown when the RHR system is not isolated from the RCS.

4. The design bases with respect to General Design Criterion 5.

5. Design bases concerned with reliability and operability requirements. The design bases regarding the manual operations required to operate the system should be described with emphasis on any operations that cannot be performed from the control room in the event of a single failure. Protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, and redundancy of instrumentation should be described. Protection against valve motor flooding and spurious single failures should be described.

6. The design bases established for the purpose of protecting the RHR system from physical damage. This discussion should include the design bases for the RHR system support structure and for protection against incidents and accidents that could render redundant components inoperative (e.g., fires, pipe whip, internally generated missiles, loss-of-coolant accident loads, seismic events).

5.4.7.2 System Design.

1. Schematic Piping and Instrumentation Diagrams. Provide a description of the RHR system, including piping and instrumentation diagrams showing all components, piping, points where connecting systems and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation. Provide a complete description of component interlocks. Provide a mode diagram showing temperatures, pressures, and flow rates for each mode of RHR operation (for example, in a BWR, the RCIC condensing mode).

2. Equipment and Component Descriptions. Describe each component of the system. Identify the significant design parameters for each component. State the design pressure and temperature of components for various portions of the system, and explain the bases for their selection. Provide pump characteristic curves and pump power requirements. Specify the available and required net positive suction head for the RHR pumps. Describe heat exchanger characteristics, including design flow rates,

143 079

inlet and outlet temperatures for the cooling fluid and for the fluid being cooled, the overall heat transfer coefficient, and the heat transfer area. Identify each component of the RHR system that is also a portion of some other system (e.g., ECCS).

3. Control. State the RHR system relief valve capacity and settings, and state the method of collection of fluids discharged through the relief valve. Describe provisions with respect to the control circuits for motor-operated isolation valves in the RHR system, including consideration of inadvertent actuation. This description should include discussions of the controls and interlocks for these values (e.g., intent of IEEE Std 279-1971), considerations for automatic valve closure (e.g., RCS pressure exceeds design pressure of residual heat removal system), valve position indications, and valve interlocks and alarms.

4. Applicable Codes and Classifications. Identify the applicable industry codes and classifications for the system design.

5. System Reliabiity Considerations. Discuss the provisions incorporated in the design to ensure that the system will operate when needed and will deliver the required flow rates (e.g., redundancy and separation of components and power sources).

6. Manual Actions. Discuss all manual actions required to be taken by an operator in order for the RHR system to operate properly with all components assumed to be operable. Identify any actions that are required to be taken from outside the control room. Repeat this discussion for the most limiting single failure in the RHR system.

5.4.7.3 Performance Evaluation. Provide an evaluation of the ability of the RHR system to reduce the temperature of the reactor coolant at a rate consistent with the design basis (5.4.7.1, item 1).

Describe the analytical methods used and clearly state all assumptions. Provide curves showing the reactor coolant temperature as a function of time for the following cases:

1. All RHR system components are operable.

2. The most limiting single failure has occurred in the RHR system.

5.4.7.4 Preoperational Testing. The proposed preoperational test program should be discussed. The discussion should identify test objectives, method of testing, and test acceptance criteria.

5.4.8 Reactor Water Cleanup System (BWRs)

This section should describe the processing capabilities and the safety-related functions of the reactor water cleanup system of a BWR.

143 080

5.4.8.1 Design Bases. The PSAR should provide the design objectives and design criteria for the reactor water cleanup system in terms of (1) maintaining reactor water purity within the guidelines of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," (2) providing system isolation capabilities to maintain the integrity of the reactor pressure boundary, and (3) precluding liquid poison removal when the poison is required for reactor shutdown. The PSAR should describe how the requirements of 10 CFR Part 50 will be implemented and should indicate the extent to which the recommendations of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and Regulatory Guide 1.29, "Seismic Design Classification," will be followed.

5.4.8.2 System Description. In the PSAR, each component should be described and its capacity provided. The processing routes and the expected and design flow rates should be indicated. Describe the instrumentation and controls provided to (1) isolate the system to maintain the reactor coolant pressure boundary, (2) isolate the system in the event the liquid poison system is needed for reactor shutdown, and (3) monitor, control, and annunciate abnormal conditions concerning the system temperature and differential pressure across filter/demineralizer units and resin strainers. Indicate the means to be used for "holding" filter/demineralizer beds intact if system flow is reduced or lost. Any control features to prevent inadvertent opening of the filter/demineralizer backwash valves during normal operation should be described. Describe the resin transfer system and indicate the provisions taken to ensure that transfers are complete and that crud traps in transfer lines are eliminated. For systems using other than filter/demineralizer units, appropriate information should be provided. The routing and termination points of system vents should be indicated. Provide piping and instrumentation diagrams indicating system interconnections and seismic and quality group interfaces. The FSAR should provide any additional information required to update the PSAR to the final design conditions.

5.4.8.3 System Evaluation. The PSAR should provide the design bases for the system capacity and should discuss the system's capability to maintain acceptable reactor water purity for normal operation, including anticipated operational occurrences (e.g., reactor startup, shutdown refueling, condensate demineralizer breakthrough, equipment downtime). Any reliance on other plant systems to meet the design objectives (e.g., liquid radwaste system) should be indicated. The design criteria for components and piping should be presented in terms of temperature, pressure, flow, or volume capacity. The seismic design and quality group classifications for components and piping should be provided. Discuss the capability of the nonregenerative heat exchanger to reduce the process temperature to a level low enough to be compatible with the cleanup demineralizer resins in the event that there is no flow return to the reactor system. The FSAR should provide any additional information required to update the PSAR to the final design conditions.

143 081

5.4.9 Main Steam Line and Feedwater Piping

5.4.10 Pressurizer

5.4.11 Pressurizer Relief Discharge System (PWR)

5.4.11.1 Design Bases. The design bases for the pressurizer relief discharge system should include the maximum step load and the consequent steam volume that the pressurizer relief tank must absorb and also the maximum heat input that the volume of water in the tank must absorb under any plant condition. This should be provided for (1) the relief valve discharge to the tank only and (2) the combined relief and safety valve discharge to the tank. The method of supporting the tank and the system should be verified.

5.4.11.2 System Description. Provide a description of the system, including the tank, the piping connections from the tank to the loop seals of the pressurizer relief and safety valves, the relief tank spray system and associated piping, the nitrogen supply piping, and the piping from the tank to the cover gas analyzer and to the reactor coolant drain tank. A piping and instrumentation diagram and a drawing of the pressurizer relief tank should be presented.

5.4.11.3 Safety Evaluation. The safety evaluation should demonstrate that the system, including the tank, is designed to handle the maximum heat load. The adequacy of the tank design pressure and temperature should be stated and justified. The results of a failure mode and effects analysis should be presented to demonstrate that the auxiliary systems serving the tank can meet the single-failure criterion without compromising safe plant shutdown. The tank rupture disk and relief valve capacities should be given, and it should be shown that their relief capacity is at least equal to the combined capacity of the pressurizer safety valves. Compliance of the system with General Design Criteria 14 and 15 should be demonstrated. The extent to which the recommendations of applicable regulatory guides such as Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," and Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," are followed should be indicated.

5.4.11.4 Instrumentation Requirements. The instrumentation and control requirements for the pressurizer relief tank and associated piping should be stated.

5.4.11.5 Inspection and Testing Requirements. The inspection and testing requirements for the pressurizer relief tank and associated piping should be described. Chapter 14 of the SAR should include a description of the preoperational and startup testing to demonstrate pressurizer relief discharge system response to step loads and transients that it is expected to accommodate during operation. Such material may be incorporated into this section by reference.

143 082

5.4.12 Valves

5.4.13 Safety and Relief Valves

5.4.14 Component Supports

143 083

6. ENGINEERED SAFETY FEATURES

Engineered safety features are provided to mitigate the consequence of postulated accidents in spite of the fact that these accidents are very unlikely. This chapter of the SAR should present information on the engineered safety features provided in the plant in sufficient detail to permit an adequate evaluation of the performance capability of these features. The information should include:

1. Descriptions of the experience, tests at simulated accident conditions, or conservative extrapolations from existing knowledge that supports the concept selection upon which the operation of the feature is based;

2. Considerations of component reliability, system interdependency, redundancy, diversity, and separation of components or portions of systems, etc., associated with ensuring that the feature will accomplish its intended purpose and will function for the period required;

3. Provisions for test, inspection, and surveillance to ensure that the feature will be dependable and effective upon demand;

4. Evidence that the material used will withstand the postulated accident environment, including radiation levels, and that radiolytic decomposition products that may occur will not interfere with it or other engineered safety features.

The engineered safety features included in plant designs vary. The engineered safety features explicitly discussed in the sections of this chapter are those that are commonly used to limit the consequences of postulated accidents in light-water-cooled power reactors. They should be treated as illustrative of the engineered safety features that should be treated in this chapter of the SAR and of the kind of informative material that is needed. Where additional or different types of engineered safety features are used, they should be covered in a similar manner in separate added sections (see Section 6.X).

This section should identify and provide a brief summary of the types of engineered safety features provided in the plant. List each system of the plant that is considered to be an engineered safety feature.

6.1 Engineered Safety Feature Materials

This section should provide a discussion of the materials used in engineered safety feature (ESF) components and the material interactions that potentially could impair operation of ESF.

143 084

6.1.1 Metallic Materials

6.1.1.1 Materials Selection and Fabrication. Information on the selection and fabrication of the materials in the engineered safety features (ESF) of the plant, such as the emergency core cooling system, the containment heat removal systems, and the containment air purification and cleanup systems should be provided. Include materials treated to enhance corrosion resistance, strength, hardness, etc. Materials for use in ESF should be selected for their compatibility with core and containment spray solutions as described in Section III of the ASME Boiler and Pressure Vessel Code, Articles NC-2160 and NC-3120.

1. List the specifications for the principal pressure-retaining ferritic materials, austenitic stainless steels, and nonferrous metals, including bolting and welding materials, in each component (e.g., vessels, piping, pumps, and valves) that is part of the ESF.

2. List the ESF construction materials that would be exposed to the core cooling water and containment sprays in the event of a loss-of-coolant accident. Show that the construction materials are compatible with the cooling and spray solutions.

3. Provide the following information to demonstrate that the integrity of the safety-related components of the ESF will be maintained during all stages of component manufacture and reactor construction:

   a. Enough details on means for avoiding significant sensitization during fabrication and assembly of austenitic stainless steel components of the ESF to demonstrate that the degree of freedom from sensitization will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

   b. Enough details on process controls for limiting exposure of austenitic stainless steel components of the ESF to contaminants capable of causing stress-corrosion cracking to show that the degree of surface cleanliness during all stages of component manufacture and reactor construction will be comparable to that obtainable by following the recommendations of Regulatory Guide 1.44 and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

   c. Details on the use of cold-worked austenitic stainless steels. If such steels have yield strengths greater than 90,000 psi, provide assurance that they will be compatible with the core cooling water and the containment sprays in the event of a loss-of-coolant accident.

   d. Enough information on the selection, procurement, testing, storage, and installation of nonmetallic thermal insulation to demonstrate that the leachable concentrations of chloride, fluoride, sodium, and silicate are comparable to the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

4. Provide enough information concerning avoidance of hot cracking (fissuring) during weld fabrication and assembly of austenitic stainless steel components of the ESF to show that the degree of weld integrity and quality will be comparable to that resulting from following the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." Describe plant requirements for welding procedures and amount and method of determination of delta ferrite in weld filler metals and in production welds, etc.

6.1.1.2 Composition, Compatibility, and Stability of Containment and Core Spray Coolants. The following information relative to the composition, compatibility, and stability of the core cooling water and the containment sprays of the ESF should be provided:

1. A description of the method used for establishing and controlling the pH of the coolants of the ESF during a loss-of-coolant accident to avoid stress-corrosion cracking of the austenitic stainless steel components and to avoid excessive generation of hydrogen by corrosion of containment metals. For all postulated design basis accidents involving release of water into the containment building, estimate the time-history of the pH of the aqueous phase in each drainage area of the building. Identify and quantify all soluble acids and bases within the containment.

2. A description of the methods used for storing ESF coolants. Demonstrate that the coolants can be stored for extended periods without significant corrosive attack on the storage vessel.

6.1.2 Organic Materials

Identify and quantify all organic materials that exist within the containment building in significant amounts. Such organic materials include wood, plastics, lubricants, paint or coatings, insulation, and asphalt. Plastics should be classified by ANSI Standard N4.1-1973, "Classification System for Polymeric Materials for Services in Ionizing Radiation" (also designated ASTM D2953-71) and paints and other coatings by Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." Coatings not intended for 40-year service without overcoating should include total coating thicknesses expected to be accumulated over the service life of the substrate surface.

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

1. Design Bases. This section should discuss the design bases for the containment, including the following information:

a. The postulated accident conditions and the extent of simultaneous occurrences (e. , seismic event, loss of offsite power, and single active failures) that determine the containment design pressure requirements (including both internal and external design pressure requirements) should be discussed. The maximum calculated accident pressure should be stated, and the bases for establishing the margin between this pressure and the design pressure should be discussed.

b. The postulated accident conditions and the extent of simultaneous occurrences (e.g., seismic event, loss of offsite power, and single active failures) that determine the design pressure requirements for the containment internal structures (i.e., containment subcompartments with reference to the design evaluation in Section 6.2.1.2) should be discussed. The maximum calculated accident pressures should be stated, and the bases for establishing the margin between this pressure and the design pressure should be discussed.

c. The postulated accident conditions and the extent of simultaneous occurrences (e.g., seismic event, loss of offsite power, and single active failures) that determine the design pressure requirements for the internal structures of pressure-suppression-type containments with reference to the design evaluation in item 3.c of this section should be discussed.

d. The sources and amounts of mass and energy that might be released into the containment and the postaccident time dependence of the mass and energy release should be discussed with reference to the design evaluations in Sections 6.2.1.3 and 6.2.1.4.

e. The effects of the engineered safety features as energy-removal systems in the containment should be discussed.

f. The capability for postaccident pressure reduction under various postulated single-failure conditions in the engineered safety feature equipment should be discussed.

g. The capability for energy removal from the containment under various postulated single-failure conditions in the engineered safety feature should be discussed.

h. The bases for establishing the containment depressurization rate should be discussed and justified with reference to the assumptions used in the analysis of the offsite radiological consequences of the accident.

i. The bases for the analysis of the minimum containment pressure used in the emergency core cooling system performance studies for PWR reactor systems should be discussed with reference to the design evaluation in Section 6.2.1.5.

143 087

j. Other design bases peculiar to pressure-suppression-type containments should be discussed with reference to the design evaluation in item 3.c of this section.

2. Design Features. This section should describe the design features of the containment structure and internal structures and should include appropriate general arrangement drawings. The following information should be included:

a. The design provisions to protect the containment structure and engineered safety feature systems against loss of function from dynamic effects (e.g., missiles and pipe whip) that could occur following postulated accidents should be discussed. Reference should be made to the detailed discussions of Chapter 3.

b. With reference to Chapter 3, the codes, standards, and guides applied in the design of the containment structure and internal structures should be identified.

c. For pressure-suppression-type containments, describe the qualification tests that are intended to demonstrate the functional capability of the structures, systems, and components (PSAR). Discuss the status of any developmental test programs that are not complete (FSAR).

d. The design provisions to protect the containment structure against loss of integrity under external pressure loading conditions resulting from inadvertent operation of containment heat removal systems or other possible modes of plant operation that could result in significant external structural loadings should be described and the functional capability of these provisions discussed. The external design pressure of the containment and the margin between the design value and the lowest expected internal pressure should be specified.

e. Identify the locations in the containment where water may be trapped and prevented from returning to the containment sump. The quantity of water involved should be specified. Discuss how the static head for recirculation pumps may be affected. Discuss the provisions that permit the water entering such regions as the refueling canal or the upper compartment of an ice condenser containment to be drained to the containment sump.

f. Discuss the functional capability and frequency of operations of the systems provided to maintain the containment and subcompartment atmospheres within prescribed pressure, temperature, and humidity limits during normal plant operation (e.g., containment penetration cooling systems, containment internal ventilation systems, and containment purge systems).

3. Design Evaluation. This section should provide evaluations of the functional capability of the containment design. The information to be included depends on the type of containment being considered (i.e.,

143 088

dry containments, ice condenser containments, or BWR water pressure-suppression-type containments) as indicated below. For new types of containment designs, information of a similar nature should be provided.

a. PWR Dry Containment (Including Subatmospheric-Type Containment). Provide analyses of the pressure response of the containment to a spectrum of postulated reactor coolant system pipe ruptures (e.g., hot leg, cold leg (pump suction), and cold leg (pump discharge) breaks). The break size and location of each postulated loss-of-coolant accident analyzed should be specified. The pressure and temperature response of the containment and the sump water temperature response as functions of time for each accident analyzed should be graphically presented up to at least 10^6 seconds after the accident, or it should be demonstrated that a lesser time includes all important aspects of the transient.

Describe the method of analysis and identify the containment computer codes used to determine the pressure and temperature response.

Refer to the mass and energy release rate data in Section 6.2.1.3 used in the analyses.

The conservatisms in the assumptions made in the analyses regarding initial containment conditions* (pressure, temperature, free volume, and humidity), containment heat removal, and emergency core cooling system operability should be discussed and demonstrated.

Provide the results of a failure mode and effects analysis of the emergency core cooling systems and containment cooling systems to determine the single active failure that maximizes the energy release to the containment and minimizes containment heat removal.

Provide the types of information described in Tables 6-1 and 6-2.

Summarize and tabulate the results of each loss-of-coolant accident analyzed as shown in Table 6-3.

Provide analyses of the temperature and pressure response of the containment to postulated secondary system pipe ruptures (e.g., steam and feedwater line breaks). The break size and location of each postulated break analyzed should be specified. Describe the method of analysis and identify the computer codes used. (Detailed mass and energy release analyses should be presented in Section 6.2.1.4.) Discuss and justify the assumptions made regarding the operating condition of the reactor, the closure times of secondary system isolation valves, and single active failures. The results of each accident analyzed should be tabulated as shown in Table 6-3.

*Best estimate at PSAR stage, more detailed listing at FSAR stage.

143 089

Provide a tabulation of the structural heat sinks within the containment in accordance with Tables 6-4A through 6-4D.* With respect to the modeling of heat sinks for heat transfer calculations, provide and justify the computer mesh spacing used for the concrete, steel, and steel-lined concrete heat sinks. Provide justification for the steel-concrete interface resistance used for the steel-lined concrete heat sinks. Provide justification for the heat transfer correlations used in the heat transfer calculations. Graphically show the condensing heat transfer coefficient as a function of time for the most severe hot leg, cold leg (pump suction), cold leg (pump discharge), and steam or feedwater line pipe breaks.

Discuss the provisions for protecting the integrity of the containment structure against the consequences of inadvertent operation of the containment heat removal systems or other systems that could result in pressures lower than the external design pressure of the containment structure. For example, if a containment vacuum relief system is provided, describe the system and show the extent to which the requirements of paragraph NE-7116 of Section III of the ASME Boiler and Pressure Vessel Code are satisfied; discuss the functional capability of the vacuum relief system. Also, discuss the administrative controls and/or electrical interlocks that would prevent such occurrences. Identify the worst single failure that could result in the inadvertent operation of the containment heat removal systems. Discuss the analytical methods and assumptions used to determine the pressure response of the containment and provide the results of analyses performed. Specify the external design pressure of the containment and setpoint for actuation of the vacuum relief system.

For the most severe reactor coolant system hot leg, cold leg (pump suction), and cold leg (pump discharge) pipe breaks, provide accident chronologies. Indicate the time of occurrence (in seconds after the break occurs) of events such as the beginning of core flood tank injection, the beginning of the ECCS injection phase, the peak containment pressure during the blowdown phase, the end of the blowdown phase, the beginning of fan-cooler operation, the beginning of the containment spray injection phase (specify the water level in the water storage tank), the peak containment pressure subsequent to the end of the blowdown phase, the end of the core reflood phase, the end of the ECCS injection phase and beginning of the recirculation phase (specify the water level in the water storage tank), the end of the containment spray injection phase (specify the water level in the water storage tank), the beginning of the containment spray recirculation phase (specify the water level in the water storage tank), the end of steam generator energy release for the post-reflood phase, and the depressurization of the containment (0 psig for subatmospheric containments, 50% of containment design pressure for conventional dry containments).

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*At the PSAR stage, the information requested may be provided on the basis of conservative estimates; however, at the FSAR stage, the information should be more definitive to complete the listing requested.

143 090

For the most severe reactor coolant system pipe breaks (i.e., the most severe pipe break in the hot leg, cold leg pump discharge, and cold leg pump suction lines) and the most severe secondary coolant system pipe break, provide energy inventories that show the distribution of energy prior to the accident, at the time of peak pressure, at the end of the blowdown phase, at the end of the core reflood phase (for loss-of-coolant accidents), and steam generator energy release during the post-reflood phase (for loss-of-coolant accidents).

The long-term performance of the containment should be described, and the capability to depressurize and maintain a low pressure (or subatmospheric pressure) within the containment should be evaluated.

Provide an evaluation of the functional capability of the normal containment ventilation system to maintain the temperature, pressure, and humidity in the containment and subcompartments within prescribed limits, assuming various single-failure conditions. Specify the limiting containment conditions for normal plant operation. Discuss the action that will be taken if these conditions are exceeded in the containment or locally, within a subcompartment.

Describe the instrumentation provided to monitor and record the containment pressure and temperature and sump temperature during the course of an accident within the containment. Discuss the range, accuracy, and response of the instrumentation and the tests conducted to qualify the instruments for use in the postaccident containment environment. Describe the recording system provided for these instruments and the accessibility of the recorders to control room personnel during a loss-of-coolant accident. Material included in Chapter 7 may be incorporated by reference.

b. Ice Condenser Containments. Provide an analysis of the pressure response of the containment to double-ended ruptures of the following high-energy lines for each control volume containing one of these lines: hot leg of reactor coolant system, cold leg of reactor coolant system, main steam line, and main feedwater line. The following information should be provided for these analyses:

(1) A graph showing the pressure response of the control volumes as functions of time for each postulated pipe break accident.

(2) A schematic diagram of the transient mass distribution (TMD) code flow network, showing all control volumes and vent flow paths used for the analysis of the particular plant design under review. Describe and justify any revisions made to the TMD code since it was reported in WCAP 8078, "Ice Condenser Containment Pressure Transient Analysis Methods," (1973).* Indicate whether the unaugmented critical flow correlation, compressibility factor "Y," and the heat transfer correlation developed

*Available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C.

143 091

from the 1974 full scale ice condenser tests reported in WCAP 8110, Supplements 6 and 7, "Test Plans and Results for the Ice Condenser System," are used in the TMD analysis.

(3) A table itemizing the volume of each control volume, the area of each vent flow path, the initial conditions for each control volume, the length of each vent flow path, the vent flow path resistances and loss coefficients, and the mass of ice, ice bed heat transfer area, and ELJAC number (condensate layer length) for each ice condenser control volume.

(4) A table comparing the maximum calculated differential pressure with the design pressure for each control volume or subcompartment. Identify the pipe break that yields the maximum calculated differential pressure for each control volume or subcompartment.

(5) The moment of inertia of the ice condenser lower inlet door, intermediate deck door, and top deck door, as well as a curve showing the flow proportioning spring force of the lower inlet door vs the door position.

(6) The types of information identified in Tables 6-2, 6-4, and 6-5, as appropriate.

Describe the ice condenser components and discuss the test programs that have been conducted to qualify the components for use in the ice condenser. If the design of components has not changed from those previously reported and accepted by the staff, the documents containing the appropriate information should be referenced. Identify all components whose designs have been changed from the design found acceptable by the staff. Describe and document the results of tests and analyses performed to qualify the new design for use in the ice condenser.

Provide an analysis of the expected reduction in the mass of ice due to sublimation during normal plant operation. Discuss the effects on the ice condenser condensing capability during a loss-of-coolant accident.

Describe the computer code (LOTIC or equivalent) used for long-term containment response analysis. Discuss and justify any changes in the mathematical models and assumptions utilized in the code relative to those utilized in previous analysis.

For the design basis accident long-term containment transient response, provide graphs from the LOTIC analysis showing the following containment parameters as functions of time:

(1) Containmer pressure

(2) Temperatures of the atmosphere in the upper and lower compartments

143 092

(3) Temperatures in the active and inactive sumps

(4) Containment spray temperature

Also provide energy distribution tables for the following events:

(1) Energy distribution at initiation

(2) End of blowdown

(3) End of reflood

(4) Completion of post-reflood steam generator energy release

(5) Completion of ice meltout

(6) Time of peak containment pressure

The tables should include the following energy sources: reactor coolant, accumulators, core stored energy, thick metal of reactor coolant system, thin metal of reactor coolant system, steam generator secondary-side fluid, and steam generator metal and the following sinks (best estimate at PSAR stage; more detailed assessment at FSAR stage): ice, structural heat sinks, spray heat exchangers, active sump, and inactive sump.

A sump model that incorporates active and inactive volumes has been utilized in the LOTIC computer code to simulate sump level and temperature history following an accident. To ensure that sufficient cooling water may be retained in the active containment sump for long-term cooling of the core and operation of the containment spray system, the following information should be provided:

(1) The capacities of the active and inactive sumps;

(2) The methods and accuracy with which the capacities of the sumps are calculated; and

(3) The time required to fill the active sump following a LOCA.

Provide analyses of the temperature and pressure response of the containment to postulated secondary system pipe ruptures (e.g., steam and feedwater line breaks). The break size and location of each postulated break analyzed should be specified. Describe the method of analysis and identify the computer codes used. (Detailed mass and energy release analyses should be presented in Section 6.2.1.4.) Discuss and justify the assumptions made regarding the operating condition of the reactor, the closure times of secondary system isolation valves, and single active failures. The results of each accident analyzed should be tabulated as shown in Table 6-3.

Discuss the manner whereby containment spray water will be returned from the upper containment to the lower compartment following a LOCA. The following information should be provided:

(1) A detailed description of the flow path by which the spray water will be able to drain back to the sump;

(2) The number and size of the drain holes;

(3) An analysis demonstrating that the drain holes are adequately sized;

(4) Drawings to show the arrangement of the drain holes; and

(5) The adminstrative control to ensure that the drain holes are open during normal operation (FSAR).

Describe the air return fan system, and provide the following information:

(1) The initiating time and the basis for sizing the air return fans;

(2) An analysis or test to demonstrate that the back draft dampers provided at the air return fan discharges have been adequately designed to withstand the dynamic force and the differential pressure across the divider deck;

(3) Fan performance curves;

(4) Analyses to show that the air return fans have sufficient head to overcome the divider barrier differential pressure;

(5) Process and instrumentation diagrams of the system; and

(6) The safety class of the system.

Describe the hydrogen skimmer system, and provide the following information:

(1) Bases and assumptions used for establishing the compartment flow rates and initiating time for the hydrogen skimmer fans;

(2) Process and instrumentation diagrams of the system;

(3) Safety class of the system;

(4) Fan performance curves; and

(5) An analysis to demonstrate that the components and ducting have been adequately designed to withstand the dynamic forces and differential pressures resulting from a LOCA.

Describe the containment vacuum relief system by providing the following information:

(1) A description of the system proposed to mitigate the consequences of inadvertent operation of the containment sprays and return air fans. Show the extent to which the requirements of paragraph NE-7116 of Section III of the ASME Boiler and Pressure Vessel Code (at least two independent relief devices) are satisfied.

(2) The worst single failure that could result in inadvertent operation of the sprays and fans.

(3) The maximum external design pressure of the containment shell.

(4) The analytical methods and assumptions used to determine the containment response to inadvertent operation of the sprays and fans.

(5) The results of analyses performed to determine the response of the containment to inadvertent operation of the sprays and fans both with and without operation of the vacuum relief system.

Describe the analytical methods and results used to establish the "external" design pressure of the internal structures (e.g., reverse pressure differentials on the operating deck and crane wall). Assumed depressurization rates in the lower compartment should be identified and justified.

Provide a table of maximum allowable operating deck bypass area as a function of reactor coolant system break size for a spectrum of break sizes up to a double-ended rupture of the largest reactor coolant system pipe. Describe the analytical methods used to determine these areas and demonstrate the conservatism in the assumptions used in the analyses. Identify all potential steam bypass leak paths and describe the design provisions taken to limit steam bypass leakage.

Discuss the potential for maldistribution of flow through the ice condenser (i.e., flow "channeling" through the ice condenser) and the effect on containment pressure response.

Discuss the design provisions made to preclude the direct impingement of a stream of fluid from high-energy lines in the lower compartment upon the ice condenser lower inlet doors.

Provide an evaluation of the functional capability of the normal containment ventilation system to maintain the temperature, pressure, and humidity in the containment and subcompartments within prescribed limits, assuming various single-failure conditions. Specify the maximum allowable containment conditions for normal plant operation. Discuss the action that will be taken if these conditions are exceeded in the containment or locally, within a subcompartment.

Provide a curve that shows the minimum containment pressure transient used in the analysis of the emergency core cooling system. Show that the containment pressure is conservatively low by describing the conservatism in the assumptions of initial containment conditions, in the modeling of the containment heat sinks, heat transfer coefficients to the heat sinks, and any other input parameter used in the containment pressure analysis. Discuss the effect of ice condenser drain water as an additional heat sink in the lower compartment and how this effect is considered in the containment pressure calculation. Identify the computer code and/or other analytical methods used to determine the minimum containment pressure transient and describe any code revisions made after the 1975 staff review of the Westinghouse ECCS evaluation model. Provide graphs showing, as functions of time, (a) the pressure, temperature, and steam condensation rates in the containment upper and lower compartments, (b) the mass and energy release rates to the containment lower compartment, (c) the containment sump temperature, and (d) the air (or vapor) flow rate between upper and lower compartments and the direction of flow.

Describe the instrumentation provided to monitor and record the containment pressure and temperature and sump temperature during the course of an accident within the containment. Discuss the range, accuracy, and response of the instrumentation and the tests conducted to qualify the instruments for use in the postaccident containment environment. Describe the recording system provided for these instruments and the accessibility of the recorders to control room personnel during a loss-of-coolant accident. Material included in Chapter 7 may be incorporated by reference.

Discuss the design provisions for monitoring the status of the ice condenser during plant operation. Discuss the ice condenser design provisions that will allow inspection and functional testing of such ice condenser components as the ice bed temperature instrumentation system; lower inlet door position monitoring system; lower, intermediate, and top deck doors; floor drains; ice condenser flow passages; divider barrier seals; refueling canal drains; and operating deck access hatches. Describe the design provisions and equipment provided to allow weighing of each ice basket.

c. <u>BWR Containments</u>. Provide the types of containment design information identified in Tables 6-6 and 6-7.

143 096

For Mark II containments, provide the results of analyses of the pressure response of the drywell and suppression chamber to a postulated rupture of the recirculation line. For Mark III containments, provide the results of analyses of the pressure response of the drywell, wetwell (that volume between the suppression pool surface and hydraulic control unit floor in the containment), and containment to postulated ruptures of the main steam line and recirculation line. Specify and justify the assumptions used in the analyses regarding the initial containment conditions, initial reactor operating conditions, energy sources, mass and energy release rates, and break areas. Graphically show the drywell pressure, wetwell pressure (Mark III), containment pressure, and deck differential pressure (Mark II) as functions of time and energy addition (e.g., blowdown, decay heat, sensible heat, pump heat) and energy removal (e.g., the RHR system, heat sinks) as a function of time.

For Mark III containments, provide the results of analyses of the pressure response of the containment and drywell to postulated ruptures of unguarded high-energy lines located in the containment. Specify and justify the assumptions used in the analyses. Describe the provisions for orificing and/or leak detection and isolation to limit the mass and energy released. Discuss the functional capability of these provisions. Graphically show the containment and drywell pressure and temperature as functions of time. Tabulate the blowdown data (time, mass flow, and enthalpy) for each pipe break analyzed.

The following tables should be provided:

(1) The initial reactor coolant system and containment conditions as identified in Table 6-8.

(2) Energy source information as identified in Table 6-9.

(3) The mass and energy release data in the format given in Table 6-10 for each pipe break accident analyzed.

(4) The information identified in Table 6-11 on the passive heat sinks* that may have been used.

(5) The results of the postulated pipe break accidents for each postulated line break in the format given in Table 6-12.

Provide the results of analyses of the transients that could lead to external pressure loads on the drywell and containment (suppression chamber). In addition, for Mark II containments provide the results of analyses of the transients that could lead to upward differential pressure loads on the drywell deck. Show that the transient used for design purposes

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*Provide best estimate of heat sink data at the PSAR stage; provide a more detailed listing of the "as built" heat sinks at the FSAR stage.

in each case is the controlling event for external pressure loading. Discuss and demonstrate the conservatism in the assumptions used in the analysis. Graphically show the containment (suppression chamber) and drywell pressures as functions of time. If a vacuum relief system is provided, describe the system and show the extent to which the requirements of paragraph NE-7116 of Section III of the ASME Boiler and Pressure Vessel Code are satisfied. Discuss the functional capability of the system. Provide the design and performance parameters for the vacuum relief devices.

Provide the results of analyses of the capability of the containment to tolerate direct steam bypass of the suppression pool for the spectrum of potential reactor coolant system break sizes. Discuss what measures are planned to minimize the potential for steam bypassing, and describe any systems provided to mitigate the consequences of steam bypass. Discuss and demonstrate the conservatism in the assumptions used in the analysis.

Describe the manner in which suppression pool dynamic loads resulting from postulated loss-of-coolant accidents, transients (e.g., relief valve actuation), and seismic events have been integrated into the affected containment structures. Provide large-size plan and section drawings of the containment illustrating all equipment and structural surfaces that could be subjected to pool dynamic loads. For each structure or group of structures, specify the dynamic loads as a function of time, and specify the relative magnitude of the pool dynamic load compared to the design basis load for each structure. Provide justification for each of the dynamic load histories by the use of appropriate experimental data and/or analyses.

Describe the manner by which potential asymmetric loads were considered in the containment design. Characterize the type and magnitude of possible asymmetric loads and the capabilities of the affected structures to withstand such a loading profile. Include consideration of seismically induced pool motion that could lead to locally deeper submergences for certain drywell to wetwell vents.

Discuss in detail the analytical models that were used to evaluate the containment and drywell responses to the postulated accidents and transients identified above. Discuss the conservatism in the models and the assumptions used. Refer to applicable test data to support the selected analytical methods. Discuss the sensitivity of the analyses to changes in key parameters.

Provide an evaluation of the functional capability of the normal containment ventilation system to maintain the temperature, pressure, and humidity in the containment and subcompartments within prescribed limits, for various assumed single-failure conditions. Specify the maximum allowable containment conditions for normal plant operation. Discuss the action that will be taken if these conditions are exceeded in the containment or locally, within a subcompartment.

143 098

Describe the instrumentation provided to monitor and record the containment pressure and temperature and sump temperature during the course of an accident within the containment. Discuss the range, accuracy, and response of the instrumentation and the tests conducted to qualify the instruments for use in the postaccident containment environment. Describe the recording system provided for these instruments and the accessibility of the recorders to control room personnel during a loss-of-coolant accident. Material included in Chapter 7 may be incorporated by reference.

6.2.1.2 Containment Subcompartments

1. Design Bases. This section should discuss the bases for the design of the containment subcompartments. The following information should be included:

a. A synopsis of the pipe break analyses performed and a justification for the selection of the design basis accident (break size and location) for each containment subcompartment,

b. The extent to which pipe restraints are used to limit the break area of pipe ruptures, and

c. The margin applied to calculated differential pressures for use in the structural design of the subcompartment walls and equipment supports.

2. Design Features. This section should provide descriptions of each subcompartment analyzed, including plan and elevation drawings showing component and equipment locations, the routing of high energy lines, and the vent locations and configurations. The subcompartment free volumes and vent areas should be tabulated (best estimate at PSAR stage, more detailed listing at FSAR stage). In addition, vent areas that become available only after the occurrence of a postulated pipe break accident (e.g., as a result of insulation collapsing or blowing out, blowout panels being blown out, or hinged doors swinging open) should be identified and the manner in which they are treated described. The availability of these vent areas should be justified. Dynamic analyses of the available vent area as a function of time should be provided and supported by appropriate test data.

3. Design Evaluation. This section should identify the computer program(s) used, and/or should present a detailed description of the analytical model, for subcompartment pressure response analyses. The results of the analyses should also be presented. The following information should be included:

a. A description of the computer program used to calculate the mass and energy release from a postulated pipe break. Provide the nodalization scheme for the system model, and specify the assumed initial operating conditions of the system. Discuss the conservatism of the blowdown model with respect to the pressure response of the subcompartment.

If the computer code being used has not been previously reviewed by the staff, provide a comparison of the blowdown to that predicted by an accepted code as justification of its acceptability.

b. The assumed initial operating conditions of the plant such as reactor power level and subcompartment pressure, temperature, and humidity.

c. A description of and justification of the subsonic and sonic flow models used in vent flow calculations. The degree of entrainment assumed for the vent mixture should also be discussed and justified.

d. The piping system within a subcompartment that is assumed to rupture, the location of the break within the subcompartment, and the break size. Give the inside diameter of the rupture of line and the location and size of any flow restrictions within the line postulated to fail.

e. The subcompartment nodalization information in accordance with the formats of Figure 6-1 and Tables 6-13 and 6-14. Demonstrate that the selected nodalization maximizes the differential pressures as a basis for establishing the design pressures for the structures and component supports.

f. Graphs of the pressure responses of all subnodes within a subcompartment as functions of time to permit evaluations of the effect on structures and component supports.

g. The mass and energy release data for the postulated pipe breaks in tabular form, with time in seconds, mass release rate in lbm/sec, enthalpy of mass released in Btu/lbm, and energy release rate in Btu/sec. A minimum of 20 data points should be used from time zero to the time of peak pressure. The mass and energy release data should be given for at least the first three seconds.

h. For all vent flow paths, the flow conditions (subsonic or sonic) up to the time of peak pressure.

i. A detailed description of the method used to determine vent loss coefficients. Provide a tabulation of the vent paths for each subcompartment and the loss coefficients.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents. This section should identify the computer codes used and/or present a detailed description of the analytical models employed to calculate the mass and energy released following a postulated loss-of-coolant accident. Various reactor coolant system pipe break locations (e.g., hot leg, cold leg pump suction, and cold leg pump discharge) and a spectrum of pipe break sizes at each location should be analyzed to ensure that the most severe pipe break location and size (i.e., the design basis loss-of-coolant accident) has been identified. The discussion should be divided into the accident phases in which different physical processes occur, as follows:

1. The blowdown phase (i.e., when the primary coolant is being rapidly injected into the containment);

2. The core reflood phase (i.e., when the core is being re-covered with water); and

3. The long-term cooling phase (i.e., when core decay heat and the remaining stored energy in the primary and secondary systems are being added to the containment).

The following information should be included:

1. Mass and Energy Release Data. For each break location, mass and energy release data should be provided for the most severe break size during the first 24 hours following the accident. (If a shorter time period is selected for some accidents, justification should be provided.) This information should be presented in tabular form, with time in seconds, mass release rate in lbm/second, and enthalpy of mass released in Btu/lbm. The table format is shown in Table 6-15. The safety injection fluid that is assumed to spill from the break directly to the containment floor should also be tabulated as a function of time.

2. Energy Sources. The sources of generated and stored energy in the reactor coolant system and secondary coolant system that are considered in analyses of loss-of-coolant accidents should be identified, and the methods used and assumptions made in calculations of the energy available for release from these sources should be described. The conservatism in the calculation of the available energy for each source should be addressed. The stored energy sources and the amounts of stored energy should be tabulated. For the sources of generated energy, curves showing the energy release rates and integrated energy released should be provided.

3. Description of Blowdown Model. The calculational procedure for determining the mass and energy released from the reactor coolant system during the blowdown phase of a loss-of-coolant accident should be described in detail or referenced as appropriate. The description should include all significant equations and correlations used in the analysis. The conservatism in the mass and energy release calculations from the standpoint of predicting the highest containment pressure response should be discussed and demonstrated. For example, calculations of the energy transferred to the primary coolant from heated surfaces and the release of primary coolant to the containment during blowdown should be described and justified. Also, the heat transfer correlations used should be presented and their application justified.

4. Description of Core Reflood Model. The calculational procedure for determining the mass and energy released to the containment during the core reflood phase of a loss-of-coolant accident should be described or referred to as appropriate. The description should include all significant equations and correlations used in the analysis. The conservatism in the mass and energy release calculations from the standpoint of predicting

143 101

the highest containment pressure response should be discussed and justified. For example, the methods of calculating the energy transferred to the emergency core cooling injection water from primary system metal surfaces and the core, the core inlet flow rate, the core exit flow rate, and the energy transferred from the steam generators should be discussed and justified. The carryout fraction used to predict the mass flow rate out of the core should be justified by comparison to experimental data such as that from the FLECHT experiments. Any assumptions made regarding the quenching of steam by ECCS injection water should be justified by comparison to appropriate experimental data. The carryout fractions, core inlet flow rate, and core inlet temperature should be provided as a function of time.

5. Description of Long-Term Cooling Model. The calculational procedure for determining the mass and energy released to the containment during the long-term cooling (or post-reflood) phase of a loss-of-coolant accident should be described or referenced as appropriate. The description should include all significant equations and correlations used in the analysis. The conservatism in the mass and energy release calculations from the standpoint of predicting the highest containment pressure response should be discussed and justified. For example, the methods of calculating (1) the core inlet and exit flow rates and (2) the removal of all sensible heat from primary system metal surfaces and the steam generators should be discussed and justified. Heat transfer correlations used should be described and their application justified. Liquid entrainment correlations for fluid leaving the core and entering the steam generators should be described and justified by comparison with experimental data. Experimental data should be provided to justify any assumptions made regarding steam quenching by ECCS water.

6. Single Failure Analysis. Provide a failure mode and effects analysis of the emergency core cooling systems to determine the single active failure that results in maximizing the energy release to the containment following a loss-of-coolant accident. This analysis should be done for each postulated break location.

7. Metal-Water Reaction. Discuss the potential for additional energy being added to the containment as a result of metal-water reaction within the core. Provide a conservative analysis of the containment pressure as a function of metal-water reaction energy addition, and demonstrate that the metal-water reaction time is conservative.

8. Energy Inventories. For the worst hot leg, cold leg pump suction, and cold leg pump discharge pipe breaks, provide inventories of the energy transferred from the primary and secondary systems to the containment and the energy remaining in the primary and secondary systems. The table format is shown in Table 6-16.

9. Additional Information Required for Confirmatory Analysis. To permit confirmatory analyses to be performed, the following information should be tabulated: the elevations, flow areas, and friction coefficients

143 102

within the primary system that are used for the containment analyses and the safety injection flow rate as a function of time. Representative values with justification should be provided for empirical correlations (such as those used to predict heat transfer and liquid entrainment) that are significant to the analysis.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment (PWR) This section should identify the computer code used and/or present a detailed description of the analytical model used to calculate the mass and energy released following a secondary system steam or feedwater line break. A spectrum of break sizes and various reactor operating conditions should be analyzed to ensure that the most severe secondary system pipe rupture has been identified. Smaller and smaller break areas of steam line breaks should be considered starting with the double-ended rupture, until no liquid entrainment is calculated to occur. The following information should be included:

1. Mass and Energy Release Data. Mass and energy release data for the most severe secondary system pipe rupture with regard to break size and location and operating power level of the reactor should be presented in tabular form with time in seconds, mass flow rate in lbm/sec, and corresponding enthalpy in Btu/lbm. Separate tables should be provided for the mass and energy released from each side of a double-ended break.

2. Single-Failure Analysis. A failure mode and effects analysis should be performed to determine the most severe single active failure for each break location for the purpose of maximizing the mass and energy released to the containment and the containment pressure response. The analysis should consider, for example, the failure of a steam or feedwater line isolation valve, the feedwater pump to trip, and containment heat removal equipment.

3. Initial Conditions. The analysis, including assumptions, to determine the fluid mass available for release into the containment should be described. In general, the analysis should be done in a manner that is conservative from a containment response standpoint (i.e., that maximizes the fluid mass available for release).

4. Description of Blowdown Model. The computer code used should be identified, and the calculational procedure should be described in detail or referenced to the appropriate topical report. All signifi t equations solved should be provided. Calculations of the energy transferred from the primary system to the secondary system, the stored energy removed from the secondary system metal, the break flow, and the steam-water separation should be conservative for containment analysis. This conservatism should be discussed and justified. The heat transfer correlations used to calculate the heat transferred from the steam generator tubes and shell should be presented and their application justified. If liquid entrainment is assumed in the break flow, appropriate experimental data should be provided.

143 103

5. Energy Inventories. For the most severe secondary system pipe rupture, inventories of the energy transferred from the primary and secondary systems to the containment should be provided. The distribution of the mass and energy released and available for release and the fluid and component temperatures within the primary and secondary systems and the containment should be given. Values should be provided for prerupture conditions, for the time of peak pressure, for the end of blowdown, and for any time a different computer code or calculational method is used in the analysis.

6. Additional Information Required for Confirmatory Analyses. To permit confirmatory analyses to be performed, the following information should be tabulated: the elevations, flow areas, and friction coefficients within the secondary system and the feedwater flow rate as a function of time. Representative values with justification should be provided for empirical correlations (such as those used to predict heat transfer and liquid entrainment) that are significant to the analysis.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System (PWR). This section should identify the computer codes used or present detailed descriptions of the analytical models used to calculate (1) the mass and energy released from the reactor coolant system following a postulated loss-of-coolant accident and (2) the containment pressure response for the purpose of determining the minimum containment pressure that should be used in analyzing the effectiveness of the emergency core cooling system. The response of the containment pressure and temperature and the sump water temperature should be plotted as functions of time. The information provided at the PSAR stage should be based on conservative values; however, as the design and construction of the facility nears completion (FSAR), more definitive data should be provided. The following information should be presented:

1. Mass and Energy Release Data. For the most severe break, state the size of the break and provide the mass and energy release data used for the minimum containment pressure analysis. This information should be presented in tabular form, with time in seconds, mass release rate in lbm/sec, and enthalpy of mass released in Btu/lbm. The quantity of safety injection fluid that is assumed to spill from the break directly to the containment floor should also be tabulated as a function of time. Discuss the conservatism in the mass and energy release analysis with regard to minimizing the containment pressure.

2. Initial Containment Internal Conditions. Specify the initial containment conditions assumed in the analysis (i.e., temperature, pressure, and humidity). Show that the initial conditions selected are conservative with respect to minimizing the containment pressure.

3. Containment Volume. Specify the assumed containment net free volume. Show that the estimated free volume of the containment has been maximized to ensure a conservative prediction of the minimum containment pressure. Discuss the uncertainty in determining the volume of the

143 104

internal structures and equipment that should be subtracted from the gross containment volume to arrive at the net free volume.

4. Active Heat Sinks. Identify the containment heat removal system and emergency core cooling system equipment that is assumed to be operative for the containment analysis. Discuss the conservatism of this assumption with respect to minimizi the containment pressure. The heat removal capacity of the engineered safeguards should be maximized by using the minimum temperature of stored water and cooling water and minimum delay times in bringing the equipment into service. Provide a figure or table showing the heat removal rate of fan cooling units as a function of containment temperature. State the containment spray flow rate and temperature assumed for the containment minimum pressure analysis. State the assumptions used in establishing the actuation times for the active heat removal systems.

5. Steam-Water Mixing. Discuss the potential for the mixing and condensation of containment steam with any spilled ECCS water during blowdown and core reflood. Comparisons with appropriate experimental data should be presented.

6. Passive Heat Sinks. With regard to the heat sink data given in Table 6-4A, 6-4B, 6-4C, and 6-4D, the uncertainty in accounting for heat sinks and in determining the heat sink parameters (such as mass, surface area, thickness, volumetric heat capacity, and thermal conductivity) should be discussed.

7. Heat Transfer To Passive Heat Sinks. The condensing heat transfer coefficients between the containment atmosphere and passive heat sinks should be discussed and justified. Comparisons with appropriate experimental data should be presented. Graphically show the condensing heat transfer coefficient as a function of time for the passive heat sinks.

8. Other Parameters. Identify any other parameters that may have a substantial effect on the minimum containment pressure analysis, and discuss how they affect the analysis. If the containment purge system is used during plant power operations, discuss the effect of a LOCA during the plant purge operation on the minimum containment pressure analysis. The radiological consequences of a LOCA during containment purge should be discussed in Chapter 15.

6.2.1.6 Testing and Inspection. This section should provide information about the containment testing and inspection program, with regard to preoperational testing and periodic inservice surveillance to ensure the functional capability of the containment and associated structures, systems, and components. Emphasis should be given to those tests and inspections considered essential to a determination that performance objectives have been achieved and performance capability is being maintained throughout the plant lifetime above preestablished limits. Such tests may include, for example, tests to determine that the ice condenser or suppression pool bypass leakage area is within allowable limits

143 105

operability tests of the air return fan system of an ice condenser containment. inspection for serviceability of the drain holes provided in the operating deck of an ice condenser containment for returning spray water in the upper compartment to the lower compartment, inspection of the ice condenser (including the condition of the ice beds and operability tests of components important to the ice condenser functional capability), and operability tests of vacuum relief systems and of mechanical devices that are required to open following a pipe break accident within a subcompartment to provide vent area. The information provided in this section should include, for example (FSAR):

1. The planned tests and inspections, including a discussion of the need and purpose of each test and inspection,

2. The selected frequency for performing each test and inspection, including justification,

3. A description of the manner in which tests and inspections will be conducted,

4. The requirements and bases for acceptability, and

5. The action to be taken in the event acceptability requirements are not met.

Particular emphasis should be given to those surveillance type tests that are of such importance to safety that they may become a part of the technical specifications of an operating license. The bases for such surveillance requirements should be discussed.

6.2.1.7 Instrumentation Requirements. This section should discuss the instrumentation to be employed for monitoring the containment conditions and actuating those systems and components having a safety function. Design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

6.2.2 Containment Heat Removal Systems.

General Design Criterion 38, "Containment Heat Removal," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires that systems to remove heat from the reactor containment be provided to rapidly reduce (consistent with the functioning of other associated systems) the containment pressure and temperature following a loss-of-coolant accident and to maintain them at acceptably low levels. General Design Criteria 39 and 40 require that the containment heat removal systems be designed to permit appropriate periodic inspection and testing to ensure the integrity and operability of the systems. The systems provided for containment heat removal include fan cooler and spray systems. The design and functional capability of these systems should be considered in this section. The design and heat removal capability of the pressure-suppression containments should be considered in Section 6.2.1.

General Design Criterion 41 requires that systems to control fission products that may be released to the containment be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environs following postulated accidents. The systems designed for containment heat removal may also possess the capability to meet this requirement. The fission product removal effectiveness of the containment heat removal systems should be considered in Section 6.5.2 of the SAR.

6.2.2.1 Design Bases. Discuss the design bases for the containment heat removal systems (i.e., the functional and mechanical and electrical design requirements of the systems). The design bases should include such considerations as:

1. The sources of energy, the energy release rates as a function of time, and the integrated energy release following postulated loss-of-coolant accidents for sizing each heat removal system;

2. The extent to which operation of the heat removal systems is relied upon to attenuate the postaccident conditions imposed on the containment (i.e., the minimum required availability of the containment heat removal systems);

3. The required containment depressurization time;

4. The capability to remain operable in the accident environment;

5. The capability to remain operable assuming a single failure;

6. The capability to withstand the Safe Shutdown Earthquake without loss of function;

7. The capability to withstand dynamic effects; and

8. The capability for periodic inspection and testing of the systems and/or system components.

6.2.2.2 System Design. Describe the design features, and provide piping and instrumentation diagrams of the containment heat removal systems. Provide a tabulation of the design and performance data for each containment heat removal system and its components.

Discuss system design requirements for redundancy and independence to ensure single-failure protection.

Discuss the system design provisions that facilitate periodic inspection and operability testing of the systems and system components.

Identify the codes, standards, and guides applied in the design of the containment neat removal systems and system components.

Specify the plant protection system signals and setpoints that actuate the containment heat removal system; alternatively, reference the section in the SAR where this information is tabulated. Provide the rationale for selecting the actuation signals and establishing the setpoints.

Specify the times following postulated accidents that the containment heat removal systems are assumed to be fully operational. Discuss the delay times following receip of the system actuation signals that are inherent in bringing the systems into service.

Discuss the extent to which the containment heat removal systems and system components are required to be remote manually operated from the main control room and the extent of operator intervention in the operation of the systems.

Describe the qualification tests that have been or will be performed on system components, such as spray nozzles, fan cooler heat exchangers, recirculation heat exchangers, pump and fan motors, valves, valve operators, and instrumentation. Discuss the test results. Demonstrate that the environmental test conditions (temperature, pressure, humidity, radiation, water pH) are representative of postaccident conditions that the equipment would be expected to be exposed to. Graphically show the environmental test conditions as a function of time or refer to the section in the SAR where this information can be found.

With respect to the fan systems, provide the following additional information:

1. Identify the ductwork and equipment housings that must remain intact following a loss-of-coolant accident;

2. Discuss the design provisions (e.g., pressure relief devices, conservative structural design) that ensure that the ductwork and equipment housings will remain intact; and

3. Provide plan and elevation drawings of the containment showing the routing of airflow guidance ductwork.

Describe the design features of the recirculation intake structures (sumps). Provide plan and elevation drawings of the structures; show the level of water in the containment following a loss-of-coolant accident in relation to the structures. Compare the design of the recirculation intake structures to the positions in Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems."

Specify the mesh size of each stage of screening and the maximum particle size that could be drawn into the recirculation piping. Of the systems that receive or may receive water from the recirculation intake structures under postaccident conditions, identify the system component that places the limiting requirement on the maximum particle size of debris that may be allowed to pass through the intake structure screening and specify the limiting particle size that the component can circulate without impairing system performance. Describe how the screening is attached to the intake structures to preclude the possibility of debris bypassing the screening.

Discuss the potential for the intake structure screening to become clogged with debris; e.g., insulation, in the light of the effective flow area of the screening and approach velocity of the water. Identify and discuss the kinds of debris that might be developed following a loss-of-coolant accident. Consider the following potential sources of debris:

1. Piping and equipment insulation,

2. Sand plug materials,

3. All structures displaced by accident pressure to provide vent area,

4. Loose insulation in the containment,

5. Debris generated by failure of non-safety-related equipment.

Describe the precautions made to minimize the potential for debris clogging the screens.

Discuss the types of insulation used inside the containment and identify where and in what quantities each type is used. List the materials of construction used for the identified insulation and describe the behavior of the insulation during and after a loss-of-coolant accident. Describe the tests performed or reference test reports available to the Commission that determined the behavior of the insulation under simulated LOCA conditions. Describe the methods of attaching the insulation to piping and components.

6.2.2.3 Design Evaluation. Describe and present the results of the spray nozzle test program to determine the drop size spectrum and mean drop size emitted from each type of nozzle as a function of pressure drop across the nozzles. Describe the analytical method employed to determine the mean spray drop size.

Provide plan and elevation drawings of the containment showing the expected spray patterns. Specify the volume of the containment covered by the sprays and the extent of overlapping of the sprays. Provide an analysis of the heat removal effectiveness of the sprays. Provide justification for the values of parameters used in the analysis (e.g., spray system flow rate as a function of time and mean spray drop size) for both full and partial spray system operation.

Graphically show the heat removal rate of the fan cooler as a function of the containment atmosphere temperature under loss-of-coolant accident conditions. Provide a figure showing the fan cooler heat removal rate as a function of the degrees of superheat for a family of curves that bound the expected containment steam-to-air ratio for the main steam line break accident. Describe the test program conducted to determine the heat removal capability of a fan cooler heat exchanger. Discuss the potential for surface fouling on the secondary side of the fan cooler heat exchanger by the cooling water and the effect on the heat removal capability of the fan cooler.

Provide analyses of the net positive suction head (NPSH) available to the recirculation pumps in accordance with the recommendations of NRC Regulatory Guide 1.1 (Safety Guide 1), "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." Provide a tabulation of the values of containment pressure head, vapor pressure head of pumped fluid, suction head, and friction head used in the analyses. Discuss the uncertainty in determining the suction head. Compare the calculated values of available NPSH for the recirculation pumps to the required NPSH of the pumps. Demonstrate the conservatism of the analyses by assuming, for the postulated loss-of-coolant accident, conditions that maximize the sump temperature and minimize the containment pressure.

Provide failure mode and effects analyses of the containment heat removal systems.

Graphically show the integrated energy content of the containment atmosphere and recirculation water as functions of time following the postulated design basis loss-of-coolant accident. Graphically show the integrated energy absorbed by the structural heat sinks and removed by the fan cooler and/or recirculation heat exchangers.

Provide an estimate of the amount of debris that could be generated during a loss-of-coolant accident and of the amount of debris to which sump inlet screens may be subjected during postulated pipe break accidents.

6.2.2.4 Tests and Inspections. Describe the program for the initial performance testing after installation and for subsequent periodic operability testing of the containment heat removal systems and system components. Discuss the scope and limitations of the tests. Describe the periodic inspection program for the systems and system components. The results of tests performed and a detailed, updated testing program should be provided in the FSAR.

6.2.2.5 Instrumentation Requirements. Describe the instrumentation provisions for actuating and monitoring the performance of the containment heat removal systems and system components. Identify the plant conditions and system operating parameters to be monitored and justify the selection of the setpoints for system actuation or alarm annunciation. Specify the locations outside the containment for instrumentation readout and alarm. The design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

143 110

6.2.3 Secondary Containment Functional Design

The secondary containment system includes the secondary containment structure and the safety-related systems provided to control the ventilation and cleanup of potentially contaminated volumes (exclusive of the primary containment) following a design basis accident. This section will discuss the secondary containment functional design. The ventilation systems (i.e., systems used to depressurize and clear the secondary containment atmosphere) should be discussed in Section 6.5.3, "Fission Product Control Systems," and Chapter 15, "Accident Analyses."

6.2.3.1 Design Bases. This section should discuss the design bases (i.e., the functional design requirements) of the secondary containment system, including the following considerations:

1. The conditions that establish the need for controlling the leakage from the primary containment structure to the secondary containment structure;

2. The functional capability of the secondary containment system to depressurize and/or maintain a negative pressure throughout the secondary containment structure and to resist the maximum potential for exfiltration under all wind loading conditions characteristic of the site;

3. The seismic design, leak tightness, and internal and external design pressures of the secondary containment structure;

4. The capability for periodic inspection and functional testing of the secondary containment structure.

6.2.3.2 System Design. Describe the design features of the secondary containment structure and provide plan and elevation drawings of the plant showing the boundary of the structure.

Provide a tabulation of the design and performance data for the secondary containment structure. Provide the types of information indicated in Table 6-17.

Discuss the performance objectives of the secondary containment structure. Identify the codes, standards, and guides applied in the design of the secondary containment structure.

Describe the valve isolation features used in support of the secondary containment. Specify the plant protection system signals that isolate and/or activate the secondary containment isolation systems or reference the section in the SAR where this information can be found.

Discuss the design provisions that prevent primary containment leakage from bypassing the secondary containment filtration systems and escaping directly to the environment. Include a tabulation of potential bypass leakage paths, including the types of information indicated in Table 6-18.

143 111

Provide an evaluation to potential bypass leakage paths considering realistic equipment design limitations and test sensitivities. The following leakage barriers in paths that do not terminate within the secondary containment should be considered potential bypass leakage paths around the leakage collection and filtration systems of the secondary containment:

1. Isolation valves in piping that penetrates both the primary and secondary containment barriers,

2. Seals and gaskets on penetrations that pass through both the primary and secondary containment barriers, and

3. Welded joints on penetrations (e.g., guard pipes) that pass through both the primary and secondary containment barriers.

Specify and justify the maximum allowable fraction of primary containment leakage that may bypass the secondary containment structure. Technical Specificatons for the identification and testing of bypass leakage paths and determination of the bypass leakage fraction should be provided in Chapter 16 of the SAR.

6.2.3.3 Design Evaluation. Provide analyses of the functional capability of the ventilation and/or cleanup systems to depressurize and/or maintain a uniform negative pressure throughout the secondary containment structure following the design basis loss-of-coolant accident. These analyses should include the effect of single active failures that could compromise the performance objective of the secondary containment system. For example, for containment purge lines that have three isolation valves in series and a leakoff valve that can be opened to the secondary containment volume between the two outboard valves, show that the failure of the outboard isolation valve to close will not prevent a negative pressure from being maintained in the secondary containment structure or result in leakage from the primary containment across the inboard valve to the environment.

If the secondary containment design leakage rate is in excess of 100%/day, an evaluation of the secondary containment system's ability to function as intended under adverse wind loading conditions characteristic of the plant site should be provided.

For analyses of the secondary containment system, provide the following information for each secondary containment volume:

1. Pressure and temperature as functions of time.

2. Primary containment wall temperature as a function of time.

3. Purge flow rate and recirculation flow rate as a function of fan differential pressure.

143 112

4. Discussion of the manner in which heat transfer from the primary containment atmosphere to the secondary containment atmosphere is calculated, including a description of the heat transfer coefficients and material properties.

5. Initial conditions assumed for the secondary containment structure and atmosphere and justification therefor.

6. Manner in which equipment heat loads within the secondary containment are considered.

7. The decrease in the secondary containment volume due to thermal and pressure expansion of the primary containment structure, and a description and justification of the methods used to calculate the volume reduction.

Identify all high-energy lines within the secondary containment structure, and provide analyses of line ruptures for any of these lines that are not provided with guard pipes.

6.2.3.4 Tests and Inspections. Describe the program for the initial performance testing and subsequent periodic functional testing of the secondary containment structures and secondary containment isolation system and system components. Discuss the scope and limitations of the tests. Describe the inspection program for the systems and system components. Results of tests performed and a detailed updated program should be provided in the FSAR. Subsequent test results should be provided as they become available.

6.2.3.5 Instrumentation Requirements. This section should describe the instrumentation to be employed for the monitoring and actuation of the ventilation and cleanup systems. Design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

6.2.4 Containment Isolation System

General Design Criteria 54, 55, 56, and 57 address design and isolation requirements for piping systems penetrating primary reactor containment. The design and functional capability of the containment isolation system should be considered in this section.

6.2.4.1 Design Bases. Discuss the bases for the design of the containment isolation system, including:

1. The governing conditions under which containment isolation becomes mandatory;

2. The criteria used to establish the isolation provisions for fluid systems penetrating the containment;

143 113

3. The criteria used to establish the isolation provisions for fluid instrument lines penetrating the containment; and

4. The design requirements for containment isolation barriers.

6.2.4.2 System Design. Provide a table of design information regarding the containment isolation provisions for fluid system lines and fluid instrument lines penetrating the containment. Include the following information in this table:

1. Containment penetration number;

2. General design criteria or regulatory guide recommendations that have been met or other defined bases for acceptability;

3. System name;

4. Fluid contained;

5. Line size (inches);

6. Engineered-safety-feature system (yes or no);

7. Through-line leakage classification (dual containments);

8. Reference to figure in SAR showing arrangement of containment isolation barriers;

9. Isolation valve number;

10. Location of valve (inside or outside containment);

11. Type C leakage test (yes or no);

12. Length of pipe from containment to outermost isolation valve (or the maximum length that will not be exceeded);

13. Valve type and operator;

14. Primary mode of valve actuation;

15. Secondary mode of valve actuation;

16. Normal valve position;

17. Shutdown valve position;

18. Postaccident valve position;

19. Power failure valve position;

143 114

20. Containment isolation signals;

21. Valve closure time; and

22. Power source.

Specify the plant protection system signals that initiate closure of the containment isolation valves or refer to the section in the SAR where this information can be found.

Provide justification for any containment isolation provisions that differ from the explicit requirements of General Design Criteria 55, 56, and 57.

Discuss the bases for the containment isolation valve closure times and, in particular, the closure times of isolation valves in system lines that can provide an open path from the containment to the environs (e.g., containment purge system).

Describe the extent to which the containment isolation provisions for fluid instrument lines meet the recommendations of Regulatory Guide 1.11 (Safety Guide 11), "Instrument Lines Penetrating Primary Reactor Containment."

Discuss the design requirements for the containment isolation barriers, including the following:

1. The extent to which the quality standards and seismic design classification of the containment isolation provisions follow the recommendations of Regulatory Guides 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and 1.29, "Seismic Design Classification;"

2. Assurance of protection against loss of function from missiles, jet forces, pipe whip, and earthquakes. Describe the provisions made to ensure that closure of the isolation valves will not be prevented by debris that could become entwined in the escaping fluid;

3. Assurance of the operability of valves and valve operators in the containment atmosphere under normal plant operating conditions and postulated accident conditions;

4. Qualification of closed systems inside and outside the containment as isolation barriers;

5. Qualification of a valve as an isolation barrier;

6. Required isolation valve closure times;

7. Mechanical and electrical redundancy to preclude common mode failures;

143 115

8. Primary and secondary modes of valve actuation.

Discuss the provisions for detecting leakage from a remote manually controlled system (such as an engineered-safety-feature system) for the purpose of determining when to isolate the affected system or system train.

Discuss the design provisions for testing the operability of the isolation valves and the leakage rate of the containment isolation barriers. Show on system drawings the design provisions for testing the leakage rate of the containment isolation barriers. Discuss the design and functional capability of associated containment isolation systems (such as isolation valve seal systems) that provide a sealing fluid or vacuum between isolation barriers and of fluid-filled systems that serve as seal systems.

Describe the environmental qualification tests that have been or will be performed on the mechanical and electrical components that may be exposed to the accident environment inside the containment. Discuss the test results. Demonstrate that the environmental test conditions (temperature, pressure, humidity, and radiation) are representative of conditions that would be expected to prevail inside the containment following an accident. Graphically show the environmental test conditions as functions of time or refer to the section in the SAR where this information can be found.

Identify the codes, standards, and guides applied in the design of the system and system components.

6.2.4.3 Design Evaluation. Provide an evaluation of the functional capability of the containment isolation system in conjunction with a failure mode and effects analysis of the system.

Provide evaluations of the functional capability of isolation valve seal systems and of fluid-filled systems that serve as seal systems.

6.2.4.4 Tests and Inspections. Describe the program for the initial functional testing and subsequent periodic operability testing of the containment isolation system and associated isolation valve seal systems if they are provided. Discuss the scope and limitations of the tests. Describe the inspection program for the isolation system and system components. The results of tests performed and a detailed updated testing and inspection program should be provided in the FSAR.

6.2.5 Combustible Gas Control in Containment

General Design Criterion 41 requires that systems be provided, as necessary, to control the concentrations of hydrogen and oxygen that may be released into the containment following postulated accidents to ensure that containment integrity is maintained.

The systems provided for combustible gas control include systems to mix the containment atmosphere, monitor combustible gas concentrations within containment regions, and reduce combustible gas concentrations within the containment. The design and functional capability of these systems should be considered in this section.

6.2.5.1 Design Bases. Discuss the bases for the design of the combustible gas control systems (i.e., the conditions under which combustible gas control may be necessary) and the functional and mechanical design requirements of the systems. The design bases should include such considerations as:

1. The generation and accumulation of combustible gases within the containment;

2. The capability to uniformly mix the containment atmosphere for as long as accident conditions require and to prevent high concentrations of combustible gases from forming locally;

3. The capability to monitor combustible gas concentrations within containment regions and to alert the operator in the main control room of the need to activate systems to reduce combustible gas concentrations;

4. The capability to prevent combustible gas concentrations within the containment from exceeding the concentration limits given in Regulatory Guide 1.7 (Safety Guide 7), "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident;"

5. The capability to remain operable, assuming a single failure;

6. The capability to withstand dynamic effects;

7. The capability to withstand the Safe Shutdown Earthquake without loss of function;

8. The capability to remain operable in the accident environment;

9. The capability to periodically inspect and test systems and/or system components;

10. The sharing of combustible gas control equipment between nuclear units at multi-unit sites;

11. The capability to transport portable hydrogen recombiner units after a loss-of-coolant accident;

12. The protection of personnel from radiation in the vicinity of the operating hydrogen recombiner units;

13. The capability to purge the containment as a backup means for combustible gas control.

143 117

6.2.5.2 System Design. Describe the design features and provide piping and instrumentation diagrams of the systems or portions of systems that comprise the combustible gas control systems and the backup purge system.

Provide a tabulation of the design and performance data for each system and its components.

Discuss system design requirements for redundancy and independence. Discuss the design provisions that facilitate periodic inspection and operability testing of the systems and system components. Identify the codes, standards, and guides applied in the design of the systems and system components.

Specify the plant protection system signals that actuate the systems and components of the combustible gas control systems and the backup purge system or refer to the section in the SAR where this information can be found.

Discuss the extent to which systems or system components are required to be manually operated from the main control room or from another point outside the containment that is accessible following an accident.

Describe the environmental qualification tests that have been or will be performed on systems (or portions thereof) and system components that may be exposed to the accident environment. Describe the test results and their applicability to the system design. Demonstrate that the environmental test conditions (temperature, pressure, humidity, and radiation) are representative of conditions that would be expected to prevail inside the containment following a loss-of-coolant accident. Graphically show the environmental test conditions as functions of time or refer to the section in the SAR where this information can be found.

With regard to the fan systems that are relied on to mix the containment atmosphere, provide the following additional information:

1. Identify the ductwork that must remain intact following a loss-of-coolant accident,

2. Discuss the design provisions (e.g., pressure relief devices, conservative structural design) that ensure that the ductwork and equipment housings will remain intact, and

3. Provide plan and elevation drawings of the containment showing the routing of the airflow guidance ductwork.

Describe the design features of the containment internal structures that promote and permit mixing of gases within the containment and subcompartments. Identify the subcompartments that are dead-ended or would not be positively ventilated following a loss-of-coolant accident and

143 118

provide analyses, assumptions, and mathematical models that ensure that combustible gases will not accumulate within them.

With regard to the system provided to continuously monitor the combustible gas concentrations within the containment following a LOCA, provide the following information:

1. A discussion of the operating principle and accuracy of the combustible gas analyzers;

2. A description of the tests conducted to demonstrate the performance capability of the analyzers or a reference to the report where such information may be found;

3. The locations of the multiple sampling points within the containment;

4. A discussion of the capability to monitor combustible gas concentrations within the containment independent of the operation of the combustible gas control systems; and

5. Failure mode and effects analyses of the containment combustible gas concentration monitoring systems.

With regard to the recombiner system provided to reduce combustible gas concentrations within the containment, provide the following additional information:

1. The operating principle of the system;

2. A description of the developmental program conducted to demonstrate the performance capability of the system and a discussion of the program results or a reference to the report where this information can be found;

3. A discussion of any differences between the recombiner system on which the qualification tests were conducted and the recombiner system that is proposed; and

4. A discussion of the extent to which equipment will be shared between nuclear power units at a multi-unit site, and the availability of the shared equipment.

<u>6.2.5.3 Design Evaluation</u>. Provide an analysis of the production and accumulation of combustible gases within the containment following a postulated loss-of-coolant accident, including the following information:

1. The assumed corrosion rate of aluminum plotted as a function of time.

143 119

2. The assumed corrosion rate of zinc plotted as a function of time.

3. An inventory of aluminum inside the containment with the mass and surface area of each item.

4. An inventory of zinc inside the containment with the total mass and surface area.

5. The mass of Zircaloy fuel cladding.

6. The quantities of hydrogen and oxygen contained in the reactor coolant system.

7. The total fission product decay power as a fraction of operating power plotted versus time after shutdown with a comparison to the decay power curve shown in Figure 6-2. Specify the reactor core thermal power rating and the assumed operating history of the reactor core.

8. The beta, gamma, and beta plus gamma energy release rates and integrated energy releases plotted as functions of time for the fission product distribution model based on the thermal power rating and operating history of the reactor core assumed in item 7 above. Indicate the extent to which the model presented in Table 1 of Regulatory Guide 1.7 is utilized.

9. The integrated production of combustible gas within the containment plotted as a function of time for each source and the concentration of combustible gas in the containment plotted as a function of time for all sources.

10. The combustible gas concentration in the containment plotted as a function of time with operation of the combustible gas reduction system assumed at full and partial capacity. Also plot the combustible gas concentration in the containment as a function of time with operation of the backup purge system assumed.

11. The basis (time or combustible gas concentrations) for activation of the combustible gas reduction and backup purge systems. Specify the design flow rates and the flow rates used in the analysis for both systems.

12. Analyses of the functional capability of the spray and/or fan systems to mix the containment atmosphere and prevent the accumulation of combustible gases within containment subcompartments. Provide plan and elevation drawings of the containment showing the airflow patterns that would be expected to result from operation of the spray and/or fan systems with a single failure assumed.

13. Analyses or test results that demonstrate the capability of the airflow guidance ductwork and equipment housings to withstand, without loss of function, the external differential pressures and internal pressure surges that may be imposed on them following a loss-of-coolant accident.

Provide failure mode and effects analyses of the combustible gas control systems.

6.2.5.4 Tests and Inspections. Describe the program for the initial performance testing and subsequent periodic operability testing of the combustible gas control systems and system components. Discuss the scope and limitations of the tests. Describe the inspection programs for the systems and system components. For those equipments that will be shared between nuclear power units at multi-unit sites, describe the program that will be conducted to ensure that the equipment can be transported within the allotted time safely and by qualified personnel. The results of tests performed and a detailed updated testing and inspection program should be provided in the FSAR.

6.2.5.5 Instrumentation Requirements. Discuss the instrumentation provisions for actuating the combustible gas control systems and backup purge system (e.g., automatically or remote manually) and monitoring the performance of the systems and system components. Identify the plant conditions and system operating parameters to be monitored and justify the selection of the setpoints for system actuation or alarm annunciation. Specify the instrumentation readout and alarm location(s) outside the containment. Design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

6.2.6 Containment Leakage Testing

General Design Criteria 52, 53, and 54 require that the reactor containment, containment penetrations, and containment isolation barriers be designed to permit periodic leakage rate testing.

Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50 specifies the leakage testing requirements for the reactor containment, containment penetrations, and containment isolation barriers.

This section should present a proposed testing program that complies with the requirements of the General Design Criteria and Appendix J to 10 CFR Part 50. All exceptions to the explicit requirements of the General Design Criteria and Appendix J should be identified and justified.

6.2.6.1 Containment Integrated Leakage Rate Test. Specify the maximum allowable containment integrated leakage rate. Describe the testing sequence for the containment structural integrity test and the containment leakage rate test.

Discuss the pretest requirements, including the requirements for inspecting the containment, taking corrective action and retesting in the event that structural deterioration of the containment is found, and reporting. Also discuss the criteria for positioning isolation valves, the manner in which isolation valves will be positioned, and the requirements for venting or draining of fluid systems prior to containment testing.

143 121

Fluid systems that will be vented or opened to the containment atmosphere during testing should be listed; the systems that will not be vented should be identified and justification given.

Describe the measures that will be taken to ensure the stabilization of containment conditions (temperature, pressure, humidity) prior to containment leakage rate testing.

Describe the test methods and procedures to be used during containment leakage rate testing, including local leakage testing methods, test equipment and facilities, period of testing, and verification of leak test accuracy.

Identify the acceptance criteria for containment leakage rate tests and for verification tests.

Discuss the provisions for additional testing in the event acceptance criteria cannot be met.

6.2.6.2 Containment Penetration Leakage Rate Test. Provide a listing of all containment penetrations. Identify the containment penetrations that are exempt from leakage rate testing and give the reasons.

Describe the test methods that will be used to determine containment penetration leakage rates. Specify the test pressure to be used.

Provide the acceptance criteria for containment penetration leakage rate testing. Specify the leakage rate limits for the containment penetrations.

6.2.6.3 Containment Isolation Valve Leakage Rate Test. Provide a listing of all containment isolation valves. Identify the containment isolation valves that are not included in the leakage rate testing and provide justification.

Describe the test methods that will be used to determine isolation valve leakage rates. Specify the test pressure to be used.

Provide the acceptance criteria for leakage rate testing of the containment isolation valves. Specify the leakage rate limits for the isolation valves.

6.2.6.4 Scheduling and Reporting of Periodic Tests. Provide the proposed schedule for performing preoperational and periodic leakage rate tests for each of the following:

1. Containment integrated leakage rate;

2. Containment penetrations; and

143 122

3. Containment isolation valves.

Describe the test reports that will be prepared and include provisions for reporting test results that fail to meet acceptance criteria.

6.2.6.5 Special Testing Requirements. Specify the maximum allowable leakage rate for the following:

1. Inleakage to subatmospheric containment, and

2. Inleakage to the secondary containment of dual containments.

Describe the test procedures for determining the above inleakage rates. Describe the leakage rate testing that will be done to determine the leakage from the primary containment that bypasses the secondary containment and other plant areas maintained at a negative pressure following a loss-of-coolant accident. Specify the maximum allowable bypass leakage.

Describe the test procedures for determining the effectiveness following postulated accidents of isolation valve seal systems and of fluid-filled systems that serve as seal systems.

6.3 Emergency Core Cooling System

6.3.1 Design Bases

A summary description of the emergency core cooling system (ECCS) should be provided. All major subsystems of the ECCS such as active high- and low-pressure safety injection systems and passive safety injection tanks should be identified. Nuclear plants that employ the same ECCS design and that are operating or have been licensed should be referenced. The purpose of the ECCS should be described and each accident or transient for which the required protection includes actuation of the ECCS should be listed.

The design bases for selecting the functional requirements for each subsystem of the ECCS should be specified. Bases for selecting such system parameters as operating pressure, ECC flow delivery rate, ECC storage capacity, boron concentration, and hydraulic flow resistance of ECCS piping and valves should be discussed.

Design bases concerned with reliability requirements should be specified. Protection against single failure in terms of piping arrangement and layout, selection of valve types and locations, redundancy of various system components, redundancy of power supplies, redundant sources of actuation signals, and redundancy of instrumentation should be described. Protection against valve motor flooding and spurious single failures should be described.

Requirements established for the purpose of protecting the ECCS from physical damage should be specified. This discussion should include design

143 123

bases for ECCS support structure design, for pipe whip protection, for missile protection, and for protection against such accident loads as loss-of-coolant accident or seismic loads.

Environmental design bases concerned with the high-temperature steam atmosphere and containment sump water level that might exist in the containment during ECCS operation should be specified.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams. Piping and instrumentation diagrams showing the location of all components, piping, storage facilities, points where connecting systems and subsystems tie together and into the reactor system, and instrumentation and controls associated with subsystem and component actuation should be provided for all modes of ECCS operation along with a complete description of component interlocks.

6.3.2.2 Equipment and Component Descriptions. Each component of the system should be described. The significant design parameters for each component should be identified. The design pressure and temperature of components for various portions of the system should be stated along with an explanation of the bases for their selection. State the quantity of coolant available (e.g., in each safety injection tank, refueling water storage tank, condensate storage tank, torus). Provide pump characteristic curves and pump power requirements. Specify the available and required net positive suction head for the ECCS pumps and identify any exceptions to the regulatory position stated in Regulatory Guide No. 1.1 (Safety Guide 1), "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." Describe heat exchanger characteristics, including design flow rates, inlet and outlet temperatures for the cooling fluid and for the fluid being cooled, the overall heat transfer coefficient, and the heat transfer area.

The relief valve capacity and settings or venting provisions included in the system should be stated. Specify design requirements for ECC delivery lag times. Describe provisions with respect to the control circuits for motor-operated isolation valves in the ECCS, including consideration of inadvertent actuation prior to or during an accident. This description should include discussions of the controls and interlocks for these valves (e.g., intent of IEEE Std 279-1971) and considerations for automatic valve closure (e.g., reactor coolant system pressure exceeds design pressure of residual heat removal system), automatic valve opening (e.g., preselected reactor coolant system pressure or ECCS signal), valve position indications, valve interlocks, and alarms.

6.3.2.3 Applicable Codes and Classifications. The applicable industry codes and classifications for the design of the system should be identified.

143 124

6.3.2.4 Material Specifications and Compatibility. Identify the material specifications for the ECCS and discuss material compatibility and chemical effects of all sorts. List the materials used in or on the ECCS by commercial name, quantity (estimate where necessary), and chemical composition. Show that the radiolytic or pyrolytic decomposition products, if any, of each material will not interfere with the safe operation of this or any other engineered safety feature.

6.3.2.5 System Reliability. Discuss the reliability considerations incorporated in the design to ensure that the system will start when needed and will deliver the required quantity of coolant within specified lag times (e.g., redundancy and separation of components, transmission lines, and power sources). Provide a failure mode and effects analysis of the ECCS. Identify the functional consequences of each possible single failure, including the effects of any single failure or operator error that causes any manually controlled electrically operated valve to move to a position that could adversely affect the ECCS. The potential for passive failures of fluid systems during long-term cooling should be considered as well as single failures of active components. For PWR plants, the single-failure analysis should consider the potential boron precipitation problem as an integral part of the requirement for providing for long-term core cooling.

Identify the specific equipment arrangement for the plant design and provide an evaluation to ensure that valve motor operators located within containment will not become submerged following a LOCA. Include all equipment in the ECCS or any other system that may be needed to limit boric acid precipitation in the reactor vessel during long-term cooling or that may be required for containment isolation.

6.3.2.6 Protection Provisions. Describe the provisions for protecting the system (including connections to the reactor coolant system or other connecting systems) against damage that might result from movement (between components within the system and connecting systems), from missiles, from thermal stresses, or from other causes (LOCA, seismic events).

6.3.2.7 Provisions for Performance Testing. The provisions to facilitate performance testing of components (e.g., bypasses around pumps, sampling lines, etc.) should be described.

6.3.2.8 Manual Actions. Identify all manual actions required to be taken by an operator in order for the ECCS to operate properly. Identify all process instrumentation available to the operator in the control room to assist in assessing postaccident conditions. Discuss the information available to the operator, the time delay during which his failure to act properly will have no unsafe consequences, and the consequences if the action is not performed at all.

6.3.3 Performance Evaluation

ECCS performance is evaluated through the safety analyses of a spectrum of postulated accidents. These analyses should be included in Chapter 15, "Accident Analyses." This section should list the accidents discussed in Chapter 15 that result in ECCS operation. The conclusions of the accident analyses should be summarized. The bases for any operational restrictions such as minimum functional capacity or testing requirements that might be appropriate for inclusion in the Technical Specifications of the license should be provided. All existing criteria that are used to judge the adequacy of ECCS performance, including those contained in § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors," of 10 CFR Part 50 should be mentioned. ECCS cooling performance evaluation should include an evaluation of single failures, potential boron precipitation (PWRs), submerged valve motors, and containment pressure assumptions (PWRs) used to evaluate the ECCS performance capability.

Simplified functional flow diagrams showing the alignment of valves, flow rates in the system, and the capacity of the ECC water supply should be provided for typical accident conditions (e.g., small- and large-break loss-of-coolant accident, steam line break). Typical flow delivery curves as a function of time should also be given for the various accidents. The time sequence of ECCS operation for short-term and long-term cooling should be discussed. Analysis supporting the selection of lag times (e.g., the period between the time an accident has occurred and the time ECC is discharged into the core) should include valve opening time, pump starting time, and other pertinent parameters. Credit for operator action should be specified.

Discuss the extent to which components or portions of the ECCS are required for operation of other systems and the extent to which components or portions of other systems are required for operation of the ECCS. An analysis of how these dependent systems would function should include system priority (which system takes preference) and conditions when various components or portions of one system function as part of another system [for example, when the water level in the reactor is below a limiting value, the recirculation pumps (i.e., residual or decay heat removal pumps) or feed pumps will supply water to the ECCS and not to the containment spray system]. Delineate any limitations on operation or maintenance included to ensure minimum capability (e.g., the storage facility common to both core cooling and containment spray systems should have provisions whereby the quantity available for core cooling will not be less than some specified quantity).

State the bounds within which principal system parameters must be maintained in the interests of constant standby readiness, e.g., such things as the minimum poison concentrations in the coolant, minimum coolant reserve in storage volumes, maximum number of inoperable components, and maximum allowable time period for which a component can be out of service.

143 126

The failure mode and effects analysis presented in Section 6.3.2.5 identifies possible degraded ECCS performances caused by single component failures. The accident analyses presented in Chapter 15 considered each of the degraded ECCS cases in the selection of the worst single failure to be analyzed. The conclusions of the various accident analyses should be discussed to show that the ECCS is adequate to perform its intended function.

6.3.4 Tests and Inspections

6.3.4.1 ECCS Performance Tests. Provide a description or reference the description of the preoperational test program performed on the ECCS. The program should provide for testing of each train of the ECCS under both ambient and simulated hot operating conditions. The tests should demonstrate that the flow rates delivered through each injection flow path using all pump combinations are within the design specifications. The adequacy of the electric power supply should be verified by testing under maximum startup loading conditions. Recirculation tests should be included in the program to demonstrate system capability to realign valves and injection pumps to recirculate coolant from the containment sump. Justify any exceptions to the regulatory position stated in Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors."

6.3.4.2 Reliability Tests and Inspections. The emergency core cooling system is a standby system that is not normally operating. Consequently, a measure of the readiness of the system to operate in the event of an accident must be achieved by tests and inspections. The periodic tests and inspections program should be identified and reasons explained as to why the program of testing planned is believed to be appropriate. The information should include:

1. Description of tests planned.

2. Considerations that led to periodic testing and the selected test frequency.

3. Test methods to be used.

4. Requirements set for acceptability of observed performance and the bases for them.

5. A description of the program for inservice inspection, including items to be inspected, accessibility requirements, and the types and frequency of inspection.

Information presented elsewhere in the SAR for the tests planned need not be repeated but only cross-referenced.

143 127

Particular emphasis should be given to those surveillance-type tests that are of such importance to safety that they may become a part of the Technical Specifications of an operating license. The bases for such surveillance requirements should be developed as a part of the SAR.

6.3.5 Instrumentation Requirements

This section should discuss the instrumentation provisions for various methods of actuation (e.g., automatic, manual, different locations). The conditions requiring system actuation together with the bases for the selection (e.g., during periods when the system is to be available, whenever the reactor coolant system pressure is less than some specified pressure, the core spray system should be actuated automatically using equipment designed to IEEE Std 279 requirements) should be included in the discussion. Design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

6.4 Habitability Systems

The term "habitability systems" refers to the equipment, supplies, and procedures provided to ensure that control room operators can remain in the control room and take actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents, as required by General Design Criterion 19 of Appendix A to 10 CFR Part 50. The habitability systems should include systems and equipment to protect the control room operators against such postulated releases as radioactive materials, toxic gases, smoke, and steam and should provide materials and facilities to permit them to remain in the control room for an extended period.

The term "control room" typically includes the main control room, areas adjacent to the main control room containing plant information and equipment that may be needed during an emergency, and kitchen and sanitary facilities. It is also the entire zone serviced by the control room ventilation system.

The habitability systems for the control room should include shielding, air purification systems, control of climatic conditions, storage capacity for food and water, and kitchen and sanitary facilities. Detailed descriptions of these systems should be included in the SAR together with evaluation of their performance. The evaluation should provide assurance that the systems will operate under all postulated conditions to permit the control room operators to remain in the control room and to take appropriate actions as required by General Design Criterion 19. Sufficient information should be provided to permit an independent evaluation of the adequacy of the systems. Information and evaluations in other sections of the SAR that relate to the adequacy of the habitability systems should be referenced (see Sections 6.5.1, 9.4.1, and 15.X.X, paragraph 5).

6.4.1 Design Basis

This section should summarize the bases on which the functional design of the habitability system and their features were established. For example, the criteria used to establish the following should be provided:

1. Control room envelope

2. Period of habitability

3. Capacity (number of people)

4. Food, water, medical supplies, and sanitary facilities

5. Radiation protection

6. Toxic or noxious gas protection

7. Respiratory, eye, and skin protection for emergencies

8. Habitability system operation during emergencies

9. Emergency monitors and control equipment

6.4.2 System Design

6.4.2.1 Definition of Control Room Envelope. The areas, equipment, and materials to which the control room operator could require access during an emergency should be identified. Those spaces requiring continuous or frequent operator occupancy should be listed. The selection of those spaces included in the control room envelope should be based on need during postulated emergencies. This information should be summarized in this section.

6.4.2.2 Ventilation System Design. This section should present the design features and fission product removal and protection capability of the control room ventilation system. Although emphasis should be placed on the emergency ventilation portion of the system, the normal ventilation system and its components also should be discussed insofar as they may affect the habitability of the control room during a design basis accident. Specifically, the following information is pertinent to the evaluation of the control room ventilation system and should be included in this section:*

1. A schematic of the control room ventilation system, including equipment, ducting, dampers, and instrumentation, and air flows for both normal and emergency modes should be noted. All dampers and valves should

*If portions of this information appear elsewhere in the SAR, they may be referenced here by section number.

be indicated with appropriate labeling (e.g., normally open or closed, manually or motor operated, fail closed or fail open).

2. A listing of major components giving their flow rates, capacities, and major design parameters. Isolation dampers should also be included in this list. Their leakage characteristics and closure times should be given.

3. The seismic classifications of components, instrumentation, and ducting. Components that are protected against missiles should be identified.

4. Layout drawings of the control room showing doors, corridors, stairwells, shielded walls, and the placement and type of equipment within the control room.

5. Elevation and plan views showing building dimensions and locations, the location of potential radiological and toxic gas releases, and the location of control room air inlets.

6. A description and placement of ventilation system controls and instruments, including the instruments that monitor the control room for radiation and toxic gases.

7. A description of the charcoal filter train, including design specifications, flow parameters, and charcoal type, weight, and distribution; HEPA filter type and specifications; and specifications of any additional components. The degree to which the recommendations of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are followed should be indicated and claimed filter efficiencies listed. (Reference may be made to Section 6.5.1.)

6.4.2.3 Leak Tightness. This section should summarize the exfiltration and infiltration analyses performed to determine unfiltered inleakage or pressurization air flow requirements. Include a listing of all potential leak paths (such as cable, pipe, and ducting penetrations, doors, dampers, construction joints, and construction materials) and their appropriate leakage characteristics. Describe the precautions and methods used to limit leakage out of or into the control room. If pressurization flow rates of less than 0.25 volume change per hour or infiltration rates of less than 0.06 volume change per hour are used, periodic leakage rate testing is normally required, and a summary of the test procedures should be included in Section 6.4.5.

6.4.2.4 Interaction With Other Zones and Pressure-Containing Equipment. A sufficiently detailed discussion should be included to indicate that the following have been taken into consideration:

143 130

1. Potential adverse interactions between the control room ventilation zone and adjacent zones that may enhance the transfer of toxic or radioactive gases into the control room. Identification should be made of any other HVAC equipment (e.g., ducts, air handling units) that may service other ventilation zones (e.g., cable spreading room, battery room) but that may be physically located within the control room habitability zone. A description should be provided of any leak paths with respect to such equipment (e.g., pilot traverse holes, hatch covers in ducts). The direction and magnitude of the pressure difference across these leak paths should be provided.

2. Isolation from the control room of all pressure-containing tanks, equipment, or piping (e.g., CO_2 firefighting containers, steam lines) that, upon failure, could cause transfer of hazardous material to the control room.

6.4.2.5 Shielding Design. Design basis accident sources of radiation other than that due to airborne contaminants within the control room should also be considered. Principal examples include fission products released to the reactor containment atmosphere, airborne radioactive contaminants surrounding the control room, and sources of radiation due to potentially contaminated equipment (e.g., control room charcoal filters and steam lines) in the vicinity of the control room. The SAR should include information describing radiation attenuation by shielding and separation. The corresponding evaluation of design basis accident doses to control room operators should be presented in Section 15.X.X, paragraph 5. Specifically, the description of the radiation shielding for the control room in a design basis accident should include the following information:

1. Accident radiation source description in terms of its origin, strength, geometry, radiation type, energy, and dose conversion factors. (Sources should include primary and secondary containments, ventilation systems, external cloud, and adjacent building air spaces.)

2. Radiation attenuation parameters (i.e., shield thickness, separation distances, and decay considerations) with respect to each source.

3. Description of potential sources of radiation streaming that may affect control room operators and the measures taken to reduce streaming to acceptable levels.

4. An isometric drawing of the control room and associated structures identifying distances and shield thicknesses with respect to each radiation source identified in 1. above.

Information pertinent to this section appearing elsewhere in the SAR should be referenced here.

6.4.3 System Operational Procedures

Discuss the method of operation during normal and emergency conditions. Discuss the automatic actions and manual procedures required to ensure effective operation of the system. If more than one emergency mode of operation is possible, indicate how the optimum mode is selected for a given condition.

6.4.4 Design Evaluations

6.4.4.1 Radiological Protection. Section 15.X.X, paragraph 5, "Radiological consequences," sets forth the documentation requirements for the evaluation of radiological exposures to plant operators from design basis accidents. The information presented in Chapter 15 should be referenced here.

6.4.4.2 Toxic Gas Protection. A hazards analysis should be performed as recommended in Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," for each toxic material identified in Section 2.2. For any of these materials that are used in the operation of the nuclear power plant, the container types and the methods of connection to the system serviced should be described. The distances between the storage locations and the air intakes to the control room should be listed along with the storage quantities. An analysis of the severity of postulated accidents involving these materials should be provided, and the steps to mitigate accident consequences should be discussed. Include descriptions of the following:

1. Principal toxic gas detector characteristics such as sensitivity, response time, principle of operation, testing and maintenance procedures, environmental qualifications, and physical location relative to the outside air intake.

2. Isolation damper transient characteristics (time to open and close) and leakage.

3. Description of the number and type of individual respiratory devices, the type of operator training for respirator use, the estimated time for deploying or donning of the equipment, the length of time the equipment can be used, and the testing and maintenance procedures.

4. Description of special ventilation system operation modes, if any, provided specifically for toxic or noxious gas conditions (e.g., bottled air pressurization, manually selected control room air purge periods).

The description of the analyses should clearly list all assumptions. Regulatory Guide 1.78 describes acceptable calculational methods. If chlorine has been identified as a potential hazard to the operator, specific guidance is provided by Regulatory Guide 1.95, "Protection of Nuclear

143 132

Power Plant Control Room Operators Against an Accidental Chlorine Release."

6.4.5 Testing and Inspection

This section should provide information about the program of testing and inspection applicable to (1) preoperational testing and (2) inservice surveillance to ensure continued integrity.

Emphasis should be given to those tests and inspections considered essential to a determination that performance objectives have been achieved and that a performance capability is being maintained above some preestablished limits throughout the plant lifetime. The information provided in this section should include, for example:

1. The planned tests and their purposes;
2. The considerations that led to the selected test frequency;
3. The test methods to be used, including a sensitivity analysis;
4. The requirements for acceptability of observed performance and the bases for them; and
5. The action to be taken if acceptability requirements are not met.

Results of any tests performed to support the specification of the test program and a detailed update of the program should be provided in the FSAR.

6.4.6 Instrumentation Requirement

This section should describe the instrumentation to be used to monitor and actuate the habitability systems. Design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

6.5 Fission Product Removal and Control Systems

This section should provide information in sufficient detail to permit the NRC staff to evaluate the performance capability of the fission product removal and control systems. Design criteria for other safety functions of the systems should be provided in other appropriate sections of this chapter. Fission product removal and control systems are considered to be those systems for which credit is taken in reducing accidental release of fission products.

The filter systems and containment spray systems for fission product removal are discussed in Sections 6.5.1 and 6.5.2, the fission product control systems in Section 6.5.3, and the ice condenser for fission product cleanup in Section 6.5.4.

6.5.1 Engineered Safety Feature (ESF) Filter Systems

All filter systems that are required to perform a safety-related function following a design basis accident should be discussed in this section. This could include filter systems internal to the primary containment, control room filters, filters on secondary confinement volumes, fuel-handling-building filters, and filters for areas containing engineered-safety-feature components. (It should be indicated in Chapter 15 which of these filters are used in mitigating the consequences of accidents.) The type of information outlined below should be provided for each of the systems. Some systems may be described in detail in other sections such as Section 9.4, but they should be listed in this section and specific reference made to the location of the information requested in each of the following sections.

6.5.1.1 Design Bases. This section should provide the design bases for each filter including the following, for example:

1. The conditions that establish the need for the filters,

2. The bases employed for sizing the filters, fans, and associated ducting, and

3. The bases for the fission product removal capability of the filters.

6.5.1.2 System Design. This section should compare the design features and fission product removal capability of each filter system to each position detailed in Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." For each ESF atmosphere cleanup system, there should be presented in tabular form a comparison between the features of the proposed system and the appropriate acceptable methods and/or characteristics presented in Regulatory Guide 1.52. For each design item for which an exception is taken, the acceptability of the proposed design should be justified in detail.

6.5.1.3 Design Evaluation. This section should provide evaluations of the filter systems to demonstrate their capability to attain the claimed filter efficiencies under the relevant accident conditions.

6.5.1.4 Tests and Inspections. Provide information concerning the program of testing and inspection applicable to preoperational testing and inservice surveillance to ensure a continued state of readiness required to reduce the radiological consequences of an accident as discussed in Regulatory Guide 1.52.

6.5.1.5 Instrumentation Requirements. Describe the instrumentation to be employed for monitoring and actuating the filter system, including the extent to which the recommendations of Regulatory Guide 1.52 are

143 134

followed. Design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

6.5.1.6 Materials. List by commercial name, quantity (estimate where necessary), and chemical composition the materials used in or on the filter system. Show that the radiolytic or pyrolytic decomposition products, if any, of each material will not interfere with the safe operation of this or any other engineered safety feature.

6.5.2 Containment Spray Systems

A detailed description of the fission product removal function of the containment spray system should be provided in this section if the system is relied on to perform this function following a design basis accident.

6.5.2.1 Design Bases. This section should provide the design bases for the fission product removal function of the containment spray system, including the following, for example:

1. The postulated accident conditions that determine the design requirements for fission product scrubbing of the containment atmosphere,

2. A list of the fission products (including the species of iodine) that the system is designed to remove and the extent to which credit is taken for the cleanup function in the analyses of the radiological consequences of the accidents discussed in Chapter 15 of the SAR, and

3. The bases employed for sizing the spray system and any components required for the execution of the atmosphere cleanup function of the system.

6.5.2.2 System Design (for Fission Product Removal). This section should provide a description of systems and components employed to carry out the fission product removal function of the spray system, including the method of additive injection (if any) and delivery to the containment. Detailed information should be provided concerning:

1. Methods and equipment used to ensure adequate delivery and mixing of the spray additive (where applicable);

2. Source of water supply during all phases of spray system operation;

3. Spray header design, including the number of nozzles per header, nozzle spacing, and nozzle orientation (a plan view of the spray headers, showing nozzle location and orientation, should be included);

4. Spray nozzle design, including information on the drop size spectrum produced by the nozzles. This information should include a histogram of the observed drop size frequency for the spatial drop size

143 135

distribution. If a mean diameter is used in the calculation of the spray effectiveness, all assumptions used for the conversion to a temporal drop size mean should be stated;

5. The operating modes of the system, including the time of system initiation, time of first additive delivery through the nozzles, length of injection period, time of initiation of recirculation (if applicable), and length of recirculation operation. Spray and spray additive flow rates should be supplied for each period of operation, assuming minimum spray operation coincident with maximum and minimum safety injection flow rates, and vice versa; and

6. The regions of the containment covered by the spray. List the containment volumes not covered by the spray and estimate the forced or convective postaccident ventilation of these unsprayed volumes. Indicate the extent to which credit is taken for the operability of ductwork, dampers, etc.

6.5.2.3 Design Evaluation. Provide an evaluation of the fission product removal function of the containment spray system. The system should be evaluated for fully effective and minimum safeguards operation, including the condition of a single failure of any active component. If the calculation of the spray effectiveness is performed for a single set of postaccident conditions, attention should be given to the effects of such parameters as temperature, spray and sump pH (and the resulting change in iodine partition), drop size, and pressure drop across the nozzle in order to ascertain whether the evaluation has been performed for a conservative set of these parameters.

6.5.2.4 Tests and Inspections. Provide a description of provisions made for testing all essential functions required for the iodine-removal effectiveness of the system. In particular, this section should include:

1. A description of the tests to be performed to verify the capability of the systems, as installed, to deliver the spray solution with the required concentration of spray additives to be used for iodine removal. If the test fluids are not the actual spray additives, describe the liquids of similar density and viscosity to be employed. Discuss the correlation of the test data with the design requirements;

2. A description of the provisions made for testing the containment spray nozzles; and

3. The provisions made for periodic testing and surveillance of any of the spray additives to verify their continued state of readiness.

Provide the bases for surveillance, test procedures, and test intervals deemed appropriate for the system.

6.5.2.5 Instrumentation Requirements. This section should include a description of any instrumentation of the spray system required for

143 136

actuation of the system and monitoring the fission product removal function of the system. Design details and logic of the instrumentation should be discussed in Chapter 7 of the SAR.

6.5.2.6 Materials. Specify and discuss the chemical composition, concentrations in storage, susceptibility to radiolytic or pyrolytic decomposition, corrosion properties, etc., of the spray additives (if any), the spray solution, and the containment sump solution.

6.5.3 Fission Product Control Systems

This section should include a detailed discussion of the operation of all fission product control systems following a design basis accident. Both anticipated and conservative operation should be described. Reference should be made to other SAR sections when appropriate. Fission product control systems are considered to be those systems whose performance controls the release of fission products following a design basis accident. These systems are exclusive of the containment isolation system and any fission product removal system, although they may operate in conjunction with fission product removal systems.

6.5.3.1 Primary Containment. This section should summarize information about the primary containment that pertains to its ability to control fission product releases following a design basis accident. This should include information such as that presented in Table 6-19. Layout drawings of the primary containment and the hydrogen purge system should be included.

Operation of containment purge systems prior to and during the accident should be discussed. Operation of the primary containment (e.g., anticipated and conservative leak rates as a function of time after initiation of the accident) should be described as applies to fission product control following a design basis accident. Where applicable, indicate when fission product removal systems are effective relative to the time sequence for operation of the primary containment following a design basis accident.

6.5.3.2 Secondary Containments. A discussion of the operation of each system used to control the release of fission products leaking from the primary containment following a design basis accident should be provided. Include the time sequence of events assumed in performing the dose estimates. Provide a table of events related to time following the design basis accident, including various parameters such as those in Table 6-2. For each time interval, indicate which fission product removal systems are effective.

Indicate both anticipated and conservative assumptions. Provide drawings that show each secondary containment volume and the ventilation system associated with that volume. Indicate the location of intake and return headers for recirculation systems and the location of exhaust intakes for once-through ventilation systems. Reference should be made to non-ESF systems that are used to control pressure in the volume.

143 137

6.5.4 Ice Condenser as a Fission Product Cleanup System

The fission product cleanup function of the ice condenser system should be considered separately from its heat removal aspects; it should be described in this section only if credit is taken for this function in the accident analyses of Chapter 15.

6.5.4.1 Design Bases. Provide the design bases for the fission product removal function of the ice condenser system, including the following, for example:

1. The postulated accident conditions and the extent of simultaneous occurrences that determine the design requirements for fission, and

2. A list of the fission products (including the species of iodine) that the system is designed to remove and the extent to which credit is taken for the cleanup function in the analyses of the radiological consequences of the accidents discussed in Chapter 15 of the SAR.

6.5.4.2 System Design (for the Fission Product Removal). This section should describe those aspects of the ice condenser design that significantly affect the fission product removal function of the ice condenser system. The information provided should include, for example:

1. The steam and air flow rates through the ice condenser as a function of time following the accident,

2. The concentrations of all additives to the ice and the pH of the ice melt and the containment sump solution following an accident, and

3. A description of the methods and equipment to be used to produce the ice with the proper additive content.

6.5.4.3 Design Evaluation. Provide an evaluation of the fission product removal function of the ice condenser system. The system should be evaluated for fully effective and minimum safeguards operation, including the condition of a single failure of any active component. If the calculation of the effectiveness is performed for a single set of post-accident conditions, attention should be given to the effects of such parameters as recirculation fan flow rate, temperature, pressure, and sump pH (and the resulting change in iodine partition) in order to ascertain that the evaluation has been performed for a conservative set of these parameters.

6.5.4.4 Tests and Inspections. Provide a description of provisions made for testing all essential functions required for the iodine-removal effectiveness of the ice condenser system and for surveillance of the system. In particular, this section should describe the provisions made for sampling the ice to verify the proper additive content.

6.5.4.5 Materials. Specify the concentrations of all additives in the ice. The effects of the additives on the long-term storage of the ice should be discussed. Address any possible reactions (e.g., slow oxidations) of the chemical additives in the ice.

6.6 Inservice Inspection of Class 2 and 3 Components

This section should discuss the inservice inspection program for Quality Group B and C components (i.e., Class 2 and 3 components in Section III of the ASME B&PV Code).

6.6.1 Components Subject to Examination

Indicate that all Quality Group B components, including those listed in Table IWC-2600 of Section XI will be examined in accordance with Code requirements. Indicate the extent to which Quality Group C components, including those listed in Subarticle IWD-2600 of Section XI, will be examined in accordance with the Code.

A detailed inservice inspection program, including information on areas subject to examination, method of examination, and extent and frequency of examination, should be provided in the technical specifications.

6.6.2 Accessibility

Indicate that the design and arrangement of Class 2 system components will provide adequate clearances to conduct the required examinations at the Code-required inspection interval, and whether the design and arrangement of Class 3 system components will also provide adequate clearances. Describe any special design arrangements made for those components that are to be examined during normal reactor operation.

6.6.3 Examination Techniques and Procedures

Indicate the extent to which the examination techniques and procedures described in Section XI of the Code will be used. Describe any special examination techniques and procedures that might be used to meet the Code requirements.

6.6.4 Inspection Intervals

Indicate that an inspection schedule for Class 2 system components will be developed in accordance with the guidance of Section XI, Subarticle IWC-2400, and whether a schedule for Class 3 system components will be developed according to Subarticle IWD-2400.

6.6.5 Examination Categories and Requirements

Indicate that the inservice inspection categories and requirements for Class 2 components are in agreement with Section XI, Subarticles IWC-2520 and IWC-2600. Indicate the extent to which inservice inspection

143 139

categories and requirements for Class 3 components are in agreement with Section XI, Subarticle IWD-2600.

6.6.6 Evaluation of Examination Results

Indicate that the evaluation of Class 2 component examination results will comply with the requirements of Article IWA-3000 of Section XI. Describe the method to be utilized in the evaluation of examination results for Class 3 components and, until the publication of IWD-3000, indicate the extent to which these methods are consistent with the requirements of Article IWA-3000 of Section XI. In addition, indicate that repair procedures for Class 2 components will comply with the requirements of Article IWC-4000 of Section XI. Describe the procedures to be utilized for repair of Class 3 components and indicate the extent to which these procedures are in agreement with Article IWD-4000 of Section XI.

6.6.7 System Pressure Tests

Indicate that the program for Class 2 system pressure testing will comply with the criteria of Code Section XI, Article IWC-5000. Indicate the extent to which the program for Class 3 system pressure tests will comply with the criteria of Article IWD-5000.

6.6.8 Augmented Inservice Inspection to Protect Against Postulated Piping Failures

Provide an augmented inservice inspection program for high-energy fluid system piping between containment isolation valves or, where no isolation valve is used inside containment, between the first rigid pipe connection to the containment penetration or the first pipe whip restraint inside containment and the outside isolation valve. This program should contain information concerning areas subject to examination, method of examination, and extent and frequency of examination.

6.7 Main Steam Line Isolation Valve Leakage Control System (BWRs)

The PSAR should describe the design bases and criteria to be applied and the preliminary system design and operation. The FSAR should describe how these requirements have been met.

6.7.1 Design Bases

This section should provide design bases for the main steam isolation valve leakage control system (MSIVLCS) in terms of:

1. The safety-related function of the system;

2. The system functional performance requirements, including the ability to function following a postulated loss of offsite power;

3. The seismic and quality group classification of the system;

143 140

4. The requirements for protection from missiles, pipe whip, and jet forces and for its ability to withstand adverse environments associated with a postulated loss-of-coolant accident (LOCA);

5. The requirements of the MSIVLCS to function following an assumed single active failure;

6. The system capabilities to provide sufficient capacity, diversity, reliability, and redundancy to perform its safety function consistent with the need for maintaining containment integrity for as long as postulated LOCA conditions require;

7. The requirements for the system to prevent or control radioactive leakage from component parts or subsystems, including methods of processing, diluting, and discharging any leakage to minimize contributing to site radioactive releases;

8. The requirements for initiation and actuation of the system consistent with the requirements for instrumentation, controls, and interlocks provided for engineered safety systems; and

9. The requirements for inspection and testing during and subsequent to power operations.

The extent to which the design guidelines of Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," will be followed should be indicated.

6.7.2 System Description

A detailed description of the MSIVLCS should be provided, including piping and instrumentation diagrams, system drawings, and location of components in the station complex. The description and drawings should also include subsystems, system operation (function), system interactions, components utilized, connection points, and instrumentation and controls utilized.

6.7.3 System Evaluation

An evaluation of the capability of the MSIVLCS to prevent or control the release of radioactivity from the main steam lines during and following a LOCA should be provided. The evaluation should include:

1. The ability of the system to maintain its safety function when subjected to missiles, pipe whip, jet forces, adverse environmental conditions, and loss of offsite power coincident with the LOCA;

2. The ability of the system to withstand the effects of a single active failure (including the failure of any one MSIV to close);

143 141

3. The protection afforded the system from the effects of failure of any non-Seismic Category I system or component;

4. The capability of the system to provide effective isolation of components and nonessential systems or equipment;

5. The capability of the system to detect and to prevent or control leakage of radioactive material to the environment. The quantity of material that could be released and the time release for each release path should be presented. (An analysis of the radiological consequences associated with the performance of this system following a design basis loss-of-coolant accident should be presented in Chapter 15.)

6. A failure mode and effects analysis to demonstrate that appropriate safety-grade instrumentation, controls, and interlocks will provide safe operating conditions, ensure system actuation following a LOCA, and preclude inadvertent system actuation; and

7. Assurance that a system malfunction or inadvertent operation will not have an adverse effect on other safety-related systems, components, or functions.

6.7.4 Instrumentation Requirements

The system instrumentation and controls should be described. The adequacy of safety-related interlocks to meet the single-failure criterion should be demonstrated.

6.7.5 Inspection and Testing

The inspection and testing requirements for the MSIVLCS should be provided. The provisions made to accomplish such inspections and testing should be described.

6.X Other Engineered Safety Features

The engineered safety features included in reactor plant designs vary from plant to plant. Accordingly, for each engineered safety feature, component, or system provided in a plant and not already referred to in this chapter of the Standard Format, the SAR should include separate sections (numbered 6.5 through 6.X) patterned after the above and providing information on:

6.X.1 Design Bases

6.X.2 System Design

6.X.3 Design Evaluation

6.X.4 Tests and Inspections

6.X.5 Instrumentation Requirements

143 142

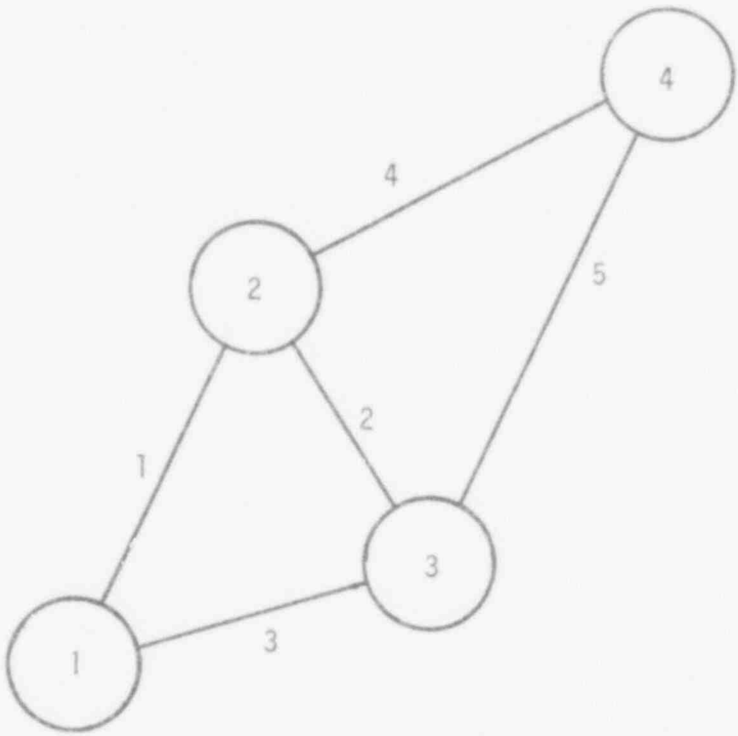
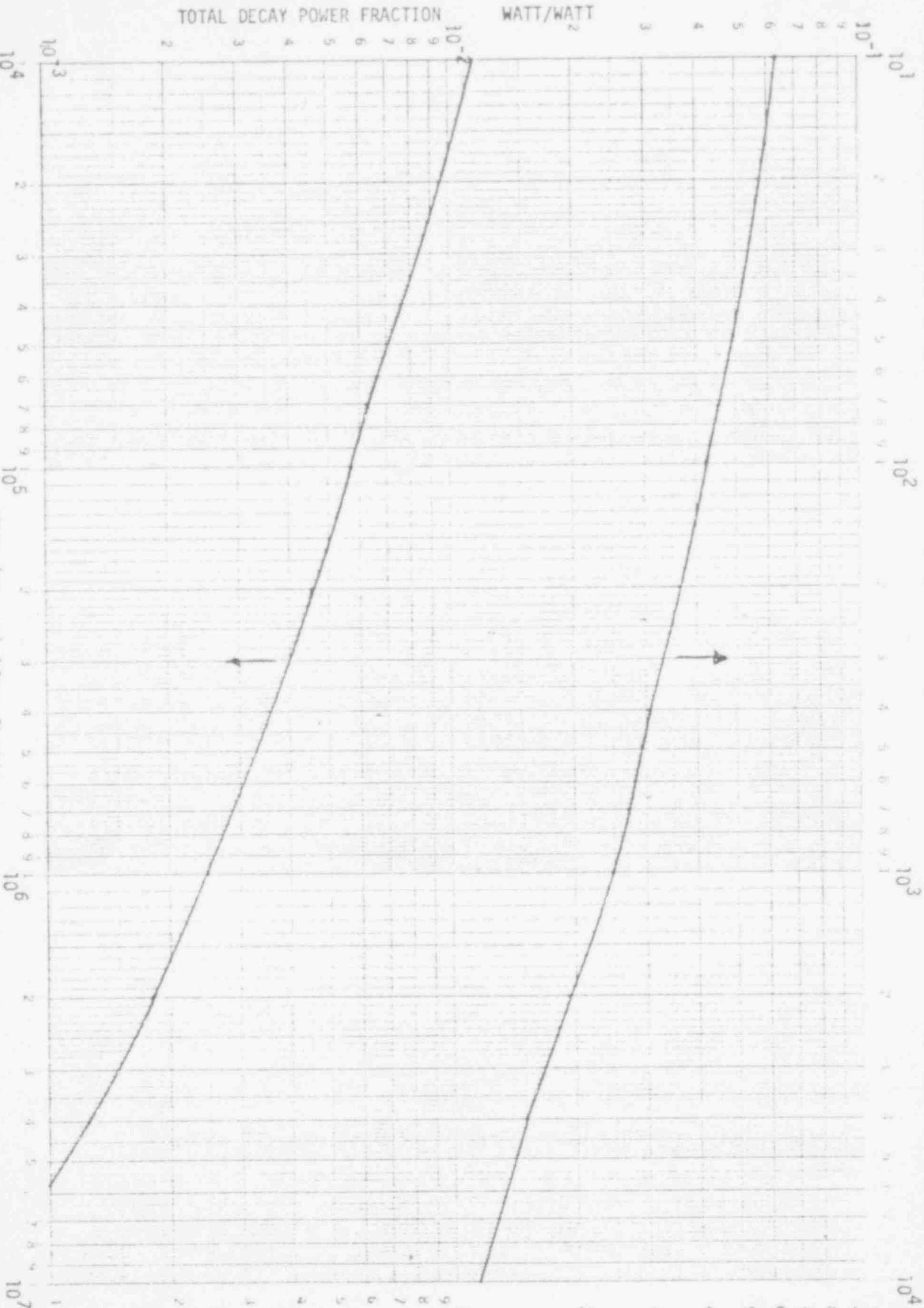


FIGURE 6-1

EXAMPLE OF SUBCOMPARTMENT NODALIZATION DIAGRAM

143 143



Time (in sec) After Shutdown

Figure 6-2

143 144

TABLE 6-1

INFORMATION TO BE PROVIDED FOR PWR DRY CONTAINMENTS
(INCLUDING SUBATMOSPHERIC CONTAINMENTS)

I. General Information

A. External Design Pressure, psig
B. Internal Design Pressure, psig
C. Design Temperature, °F
D. Free Volume, ft^3
E. Design Leak Rate, %/day @ psig

II. Initial Conditions

A. Reactor Coolant System (at design overpower of 102% and at normal liquid levels)

1. Reactor Power Level, MWt
2. Average Coolant Temperature, °F
3. Mass of Reactor Coolant System Liquid, lbm
4. Mass of Reactor Coolant System Steam, lbm
5. Liquid plus Steam Energy,* Btu

B. Containment

1. Pressure, psig
2. Temperature, °F
3. Relative Humidity, %
4. Service Water Temperature, °F
5. Refueling Water Temperature, °F
6. Outside Temperature, °F

C. Stored Water (as applicable)

1. Borated-Water Storage Tank, ft^3
2. All Accumulators (safety injection tanks), ft^3
3. Condensate Storage Tanks, ft^3

*All energies are relative to 32°F.

143 145

TABLE 6-2

PWR ENGINEERED SAFETY FEATURE SYSTEMS INFORMATION

As indicated below, this information should be provided for two conditions: (1) full-capacity operation and (2) the capacities used in the containment analysis.

	Full Capacity	Value Used for Containment Analysis
A. Passive Safety Injection System		
1. Number of Accumulators (Safety Injection Tanks)		
2. Pressure Setpoint, psig		
B. Active Safety Injection Systems		
1. High-Pressure Safety Injection		
a. Number of Lines		
b. Number of Pumps		
c. Flow Rate, gpm		
2. Low-Pressure Safety Injection		
a. Number of Lines		
b. Number of Pumps		
c. Flow Rate, gpm		
C. Containment Spray System		
1. Injection Spray		
a. Number of Lines		
b. Number of Pumps		
c. Number of Headers		
d. Flow Rate, gpm		
2. Recirculation Spray		
a. Number of Lines		
b. Number of Pumps		

143 146

TABLE 6-2 (Continued)

	Full Capacity	Value Used for Containment Analysis
c. Number of Headers		
d. Flow Rate, gpm		
D. Containment Fan Cooler System		
1. Number of Units		
2. Air-Side Flow Rate, cfm		
3. Heat Removal Rate at Design Temperature, 10^6 Btu/hr		
4. Overall Heat Transfer Coefficient, Btu/hr-ft^2-°F		
E. Heat Exchangers		
1. Recirculation Systems		
a. System		
b. Type		
c. Number		
d. Heat Transfer Area, ft^2		
e. Overall Heat Transfer Coefficient, Btu/hr-ft^2-°F		
f. Flow Rates:		
(1) Recirculation Side, gpm		
(2) Exterior Side, gpm		
g. Source of Cooling Water		
h. Flow Begins, sec		
F. Others		

143 147

TABLE 6-3

SUMMARY OF CALCULATED CONTAINMENT PRESSURE AND TEMPERATURES

Pipe Break Location and Break Area, ft^2	Calculated Value
Peak Pressure, psig	
Peak Temperature, °F	
Time of Peak Pressure, sec	
Energy Released to Containment up to the End of Blowdown, 10^6 Btu	

143 148

TABLE 6-4

PASSIVE HEAT SINKS

A. LISTING OF PASSIVE HEAT SINKS*

The following structures, components, and equipment are examples of passive heat sinks that should be included in the submittal, as appropriate:

Containment Building

1. Building/liner
2. External concrete walls
3. Building liner steel anchors
4. Building floor and sump
5. Personnel hatches
6. Equipment hatches

Internal Structures

7. Internal separation walls and floors
8. Refueling pool and fuel transfer pit walls and floors
9. Crane wall
10. Primary shield walls
11. Secondary shield walls
12. Piping tunnel
13. Pressurizer room
14. Reheat exchanger room
15. Valve room
16. Fuel canal shielding
17. Jet impingement deflectors
18. Regenerative heat exchanger shield
19. Other

Lifting Devices

20. Lifting rig
21. Refueling machine
22. Vessel head lifting rig
23. Polar crane
24. Manipulator crane
25. Other

Supports

26. Reactor vessel supports
27. Steam generator supports

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* Provide best estimates of these heat sinks in the PSAR stage and a detailed listing in the FSAR.

143 149

TABLE 6-4 (Continued)

28. Fuel canal support
29. Reactor coolant pump supports
30. Safety injection tank supports
31. Pressure relief tank supports
32. Drain tank supports
33. Fan cooler support
34. Other

Storage Racks

35. Fuel storage
36. Head storage
37. Other

Gratings, Ladders, etc.

38. Ladders, stairways
39. Floor plates
40. Steel handrails and platform railings
41. Steel gratings
42. Steel risers
43. Steel tread and stringers

Electrical Equipment

44. Cables, conduits
45. Cable trays
46. Instrumentation and control equipment, electrical boxes
47. Electric penetrations

Piping Support Equipment

48. Restraints
49. Hangers
50. Piping penetrations

Components

51. Reactor heat removal pumps and motors
52. Reactor coolant pump motors
53. Hydrogen recombiners
54. Fan coolers
55. Reactor cavity and support cooling units
56. Air filter units
57. Air blowers
58. Air heating equipment

143 150

TABLE 6-4 (Continued)

59. Safety injection tanks
60. Pressurizer quench tank
61. Reactor drain tank
62. Other

Uninsulated Cold-Water-Filled Piping and Fittings

63. Reactor heat removal system
64. Service water system
65. Component cooling water system
66. Other

Drained Piping and Fittings

67. Containment spray piping and headers
68. Other

Heating, Ventilation, and Air Conditioning

69. Ducting
70. Duct dampers

143 151

TABLE 6-4 PASSIVE HEAT SINKS (Continued)

B. MODELING OF PASSIVE HEAT SINKS

The following data should be provided for the passive heat sinks listed in Table 6-4A (best estimates in the PSAR stage and a detailed listing in the FSAR stage):

		Unpainted Material					
		Metal			Concrete Exposed Surface Area By Thickness Group,* ft²		
Passive Heat Sink	Painted Material Thickness ft	Material	Exposed Surface Area By Thickness Group* 1 2 ...6, ft²	Total Mass, lb	a	b	Total Surface ft²
1. Vessel steel plate							
2. External concrete walls							
3. Vessel liner steel anchors							
TOTALS							
Painted Surfaces							
Unpainted Surfaces							

* See Table 6-4C

TABLE 6-4 PASSIVE HEAT SINKS (Continued)

C. THICKNESS GROUPS

Material	Group Designation	Thickness Range, in.
Metal	1	0-0.125
	2	0.125-0.25
	3	0.25-0.5
	4	0.50-1.00
	5	1.00-2.50
	6	>2.50
Concrete	a	0-3.0
	b	>3.0

143 153

TABLE 6-4 PASSIVE HEAT SINKS (Continued)

D. THERMOPHYSICAL PROPERTIES OF PASSIVE HEAT SINK MATERIALS

Material	Density, lb/ft^3	Specific Heat, Btu/lb-°F	Thermal Conductivity, Btu/hr-ft-°F

143 154

TABLE 6-5

INFORMATION TO BE PROVIDED FOR ICE CONDENSER CONTAINMENTS

I. Lower Compartment

- A. Free Volume, ft^3
- B. Design Pressure, psig
- C. Design Temperature, °F
- D. Peak Pressure, DBA, psig
- E. Pressure Margin, %
- F. Normal Operating Temperature, °F
- G. Normal Operating Pressure, psia
- H. Normal Operating Relative Humidity, %

II. Upper Compartment

- A. Free Volume, ft
- B. Design Pressure, psig
- C. Design Temperature, °F
- D. Peak Pressure, DBA, psig
- E. Pressure Margin, %
- F. Normal Operating Temperature, °F
- G. Normal Operating Pressure, psia
- H. Normal Operating Relative Humidity, %

III. Ice Condenser

- A Ice Weight, lb
- B. Flow Area, ft^2
- C. Length/Hydraulic Diameter
- D. Channel Surface Area, ft^2
- E. Ice Basket Diameter, ft
- F. Inlet Door Area, ft^2
- G. Ice Condenser Flow Area, ft^2
- H. Volume, ft^3
- I. Ice Bed Height, ft
- J. Inlet Door Opening Pressure, psf
- K. Ice Boron Concentration, ppm
- L. O.D., ft
- M. I.D., ft

IV. Refrigeration Cooling Capacity

- A. Cooling Capacity for Compartment, tons
- B Number of Fan Coolers per Unit
- C. Air Temperature to Insulated Panels, °F

TABLE 6-5 (Continued)

V. General Information

A. External Design Pressure, psig
B. Internal Design Pressure, psig
C. Design Leak Rate, %/day @ psig

VI. Initial Conditions

A. Reactor Coolant System (at design overpower of 102% and at normal liquid levels)

1. Reactor Power Level, MWt
2. Average Coolant Temperature, °F
3. Mass of Reactor Coolant System Liquid, lbm
4. Mass of Reactor Coolant System Steam, lbm
5. Liquid plus Steam Energy,* Btu

B. Containment

1. Pressure, psig
2. Temperature, °F (upper compartment, lower compartment, and ice condenser)
3. Relative Humidity, % (upper compartment, lower compartment, and ice condenser)
4. Service Water Temperature, °F
5. Refueling Water Temperature, °F
6. Outside Temperature, °F

C. Stored Water (as applicable)

1. Borated Water Storage Tank, ft^3
2. All Accumulators (safety injection tanks), ft^3
3. Condensate Storage Tanks, ft^3

* All energies are relative to 32°F.

143 156

TABLE 6-6

INFORMATION TO BE PROVIDED FOR WATER POOL PRESSURE-SUPPRESSION CONTAINMENTS

A. Drywell

1. Internal Design Pressure, psig (Mark II)
2. Drywell Deck Design Differential Pressure, psid (Mark II)
3. Drywell Design Differential Pressure, psid (Mark III)
4. External Design Pressure, psig
5. Design Temperature, °F
6. Free Volume, ft^3
7. Design Leak Rate, %/day @ psig

B. Containment (Wetwell)

1. Internal Design Pressure, psig
2. External Design Pressure, psig
3. Design Temperature, °F
4. Air Volume (min/max), ft^3
5. Wetwell Air Volume, ft^3 (Mark III)
6. Pool Volume (min/max), ft^3
7. Suppression Pool Makeup Volume, ft^3 (Mark III)
8. Pool Surface Area, ft^2
9. Pool Depth (min/max), ft
10. Design Leak Rate, %/day @ psig
11. Hydraulic Control Unit Floor Flow Restriction, % restricted (Mark III)

C. Vent System

1. Number of Vents
2. Vent Diameter, ft
3. Net Free Vent Area, ft^2
4. Vent Submergence(s) (min/max), ft
5. Vent System Loss Factors
6. Drywell Wall to Weir Wall Distance, ft (Mark III)
7. Net Weir Annulus Cross-Sectional Area, ft^2 (Mark III)

143 157

TABLE 6-7

ENGINEERED SAFETY FEATURE SYSTEMS INFORMATION FOR WATER-POOL PRESSURE-SUPPRESSION CONTAINMENTS

This information should be provided for two conditions: (1) full-capacity operation and (2) the capacities used in the containment analysis.

A. Containment Spray System

1. Number of Spray Pumps
2. Capacity per Pump, gpm
3. Number of Spray Headers
4. Spray Flow Rate - Drywell, lb/hr
5. Spray Flow Rate - Wetwell, lb/hr
6. Spray Thermal Efficiency, %

B. Containment Cooling System

1. Number of Pumps
2. Capacity per Pump, gpm
3. Number of Heat Exchangers
4. Heat Exchanger Type
5. Heat Transfer Area per Exchanger, ft^2
6. Overall Heat-Transfer Coefficient, Btu/hr ft^2 ^oF
7. Secondary Coolant Flow Rate per Exchanger, lb/hr
8. Design Service Water Temperature (min/max), ^oF

143 158

TABLE 6-8

INITIAL CONDITIONS FOR ANALYSIS OF WATER-POOL PRESSURE-SUPPRESSION CONTAINMENTS

A. Reactor Coolant System (at design overpower of 102% and at normal liquid levels)

1. Reactor Power Level, MWt
2. Average Coolant Pressure, psig
3. Average Coolant Temperature, °F
4. Mass of Reactor Coolant System Liquid, lb
5. Mass of Reactor Coolant System Steam, lb
6. Volume of Water in Reactor Vessel, ft^3
7. Volume of Steam in Reactor Vessel, ft^3
8. Volume of Water in Recirculation Loops, ft^3

B. Drywell

1. Pressure, psig
2. Temperature, °F
3. Relative Humidity, %

C. Containment (suppression chamber)

1. Pressure, psig
2. Air Temperature, °F
3. Water Temperature, °F
4. Relative Humidity, %
5. Water Volume, ft^3
6. Vent Submergence, ft

TABLE 6-9

ENERGY SOURCES FOR WATER-POOL PRESSURE-SUPPRESSION CONTAINMENT ACCIDENT ANALYSES

1. Decay heat rate, Btu/sec, as a function of time
2. Primary system sensible heat release to containment, Btu/sec, as a function of time
3. Metal-water reaction heat rate, Btu/sec, as a function of time
4. Heat release rate from other sources, Btu/sec, as a function of time

143 160

TABLE 6-10

MASS AND ENERGY RELEASE DATA FOR ANALYSIS OF WATER-POOL PRESSURE-SUPPRESSION CONTAINMENT ACCIDENTS

A. Recirculation Line Break

1. Pipe I.D., in.
2. Effective Total Break Area, ft^2, versus time
3. Name of Blowdown Code
4. Blowdown Table

Time, sec	Flow, lb/sec	Enthalpy, Btu/lb	Reactor Vessel Pressure, psig
0			
t_1			
t_2			
t_n		-BLOWDOWN COMPLETED-	

B. Main Steam Line Break

1. Pipe I.D., in.
2. Effective Total Break Area, ft^2, versus time
3. Name of Blowdown Code
4. Blowdown Table

Time, sec	Flow, lb/sec	Enthalpy, Btu/lb	Reactor Vessel Pressure, psig
0			
t_1			
t_2			
.			
.			
t_n			

TABLE 6-11

PASSIVE HEAT SINKS USED IN THE ANALYSIS OF BWR PRESSURE-SUPPRESSION CONTAINMENTS
(If Applicable)

A. Listing of Passive Heat Sinks

Provide a listing of all structures, components, and equipment used as passive heat sinks (see Table 6-4A).

B. Detailed Passive Heat Sink Data

The information to be provided and the format are given in Table 6-4B, 6-4C, and 6-4D.

C. Heat Transfer Coefficients

Graphically show the condensing heat transfer coefficients as functions of time for the design basis accident.

143 162

TABLE 6-12

RESULTS OF WATER-POOL PRESSURE-SUPPRESSION CONTAINMENT ACCIDENT ANALYSES

The information presented below should be based on the values used for containment analysis presented in Table 6-7.

A. Accident Parameters

	Recirculation Line Break	Steam Line Break
1. Peak Drywell Pressure, psig (Mark II)		
2. Peak Drywell Deck Differential Pressure, psid (Mark II)		
3. Peak Drywell Differential Pressure, psid (Mark III)		
4. Time(s) of Peak Pressures, sec		
5. Peak Drywell Temperature, °F		
6. Peak Containment (Suppression Chamber) Pressure, psig		
7. Time of Peak Containment Pressure, sec		
8. Peak Wetwell Pressure, psig		
9. Time of Peak Wetwell Pressure, sec		
10. Peak Containment Atmospheric Temperature, °F		
11. Peak Suppression Pool Temperature, °F		

The above tabulation should be supplemented by plots of containment and drywell pressure and temperature, vent flow rate, energy release rate, and energy removal rate as functions of time to at least 106 seconds.

TABLE 6-12 (Continued)

B. Energy Balance of Sources and Sinks

	Time, sec			
	Initial	Drywell Peak Pressure	End of Blowdown	Long-Term Peak Pressure
	0			
		Energy, 10^6 Btu		
1. Reactor Coolant				
2. Fuel and Cladding				
3. Core Internals				
4. Reactor Vessel Metal				
5. Reactor Coolant System Piping, Pumps, and Valves				
6. Blowdown Enthalpy				
7. Decay Heat				
8. Metal-Water Reaction Heat				
9. Drywell Structures				
10. Drywell Air				
11. Drywell Steam				
12. Containment Air				
13. Containment Steam				
14. Suppression Pool Water				
15. Heat Transferred by Heat Exchangers				
16. Passive Heat Sinks				

143 164

TABLE 6-13

SUBCOMPARTMENT VENT PATH DESCRIPTION

VENT PATH NO.	FROM VOL. NODE NO.	TO VOL. NODE NO.	DESCRIPTION OF VENT PATH FLOW		AREA ft^2	LENGTH ft	HYDRAULIC DIAMETER ft	HEAD LOSS, K				
			CHOKED	UNCHOCKED				FRICTION K, ft/d	TURNING LOSS, K	EXPANSION, K	CONTRACTION, K	TOTAL

TABLE 6-14

SUBCOMPARTMENT NODAL DESCRIPTION

				INITIAL CONDITIONS			DBA BREAK CONDITIONS						
VOLUME NO.	DESCRIPTION	HEIGHT, ft	CROSS-SECTIONAL AREA, ft^2	TEMP. °F	PRESS. psia	HUMID. %	BREAK LOC. VOL. NO.	BREAK LINE	BREAK AREA ft^2	BREAK TYPE	CALC. PEAK PRESS DIFF. psig	DESIGN PEAK PRESS DIFF. psig	DESIGN MARGIN, %

143 166

TABLE 6-15

MASS AND ENERGY RELEASE RATE DATA
FOR POSTULATED LOSS-OF-COOLANT ACCIDENTS

Pipe I.D., in.

Break Area, ft^2

Time, sec	Mass Release Rate, lbm/sec	Enthalpy, Btu/lbm	Reactor Vessel Pressure, psig
0			
t_1			
t_2	Blowdown Phase		
.			
.			
.			
t End of Blowdown			
.			
.			
.	Core Reflood Phase		
t End of Core Reflood			
.			
.			
.			
t End of Post-Reflood	Post-Reflood Phase		
.			
.			
.			
. End of Problem	Post-Post-Reflood (or Decay Heat) Phase		

143 167

143 168

TABLE 6-16

REACTOR CONTAINMENT BUILDING ENERGY DISTRIBUTION
PIPE BREAK LOCATION AND PIPE BREAK AREA

Note: The datum temperature is 32°F unless otherwise noted.

	Energy, 10^6 Btu					
	Prior to LOCA	At Peak Pressure Prior to End of Blowdown	End of Blowdown	At Peak Pressure after End of Blowdown	End of Core Reflood	One Day into Recirc.
Reactor Coolant Internal Energy						
Core Flood Tank Coolant Internal Energy						
Energy Stored in Core						
Energy Stored in RV Internals						
Energy Stored in RV Metal						
Energy Generated During Shutdown from Decay Heat						
Energy Stored in Pressurizer, Primary Piping, Valves, and Pumps						
Energy Stored in Steam Generator Metal						
Secondary Coolant Internal Energy (in Steam Generators)						
Energy Content of RCB Atmosphere *						

TABLE 6-16 (Continued)

	Energy, 10^6 Btu					
	Prior to LOCA	At Peak Pressure Prior to End of Blowdown	End of Blowdown	At Peak Pressure after End of Blowdown	End of Core Reflood	One Day into Recirc.
Energy Content of RCB and Internal Structures **						
Energy Content of Recirculation Intake Water						
Energy Content of BWST W[illegible]						
[illegible] Removed by Decay [illegible] Coolers						
En[illegible] by Re[illegible] Containment Building Fan Coolers						

* Atmospheric constituent datums are 120°F for air and 32°F for water vapor.
** Datum for energy content of Reactor Containment Building and internal structures is 120°F.

143 169

TABLE 6-17

ADDITIONAL INFORMATION TO BE PROVIDED FOR DUAL-CONTAINMENT PLANTS

I. Secondary Containment Design

For each volume comprising the secondary containment, provide the following information:

A. Free Volume, ft^3

B. Pressure, inches of water, gauge

1. Normal Operation

2. Postaccident

C. Leak Rate at Postaccident Pressure (%/day)

D. Exhaust Fans

1. Number

2. Type

E. Filters

1. Number

2. Type

II. Transient Analysis

A. Initial Conditions (provide for each volume if different)

1. Pressure, psia

2. Temperature, °F

3. Outside Air Temperature, °F

4. Thickness of Secondary Containment Wall, in

5. Thickness of Primary Containment Wall, in

143 170

TABLE 6-17 (Continued)

B. Thermal Characteristics

1. Primary Containment Wall

   a. Coefficient of Linear Expansion, in/in-°F (if applicable)
   b. Modulus of Elasticity, psi (if applicable)
   c. Thermal Conductivity, Btu/hr-ft-°F
   d. Thermal Capacitance, Btu/ft^3-°F

2. Secondary Containment Wall

   a. Thermal Conductivity, Btu/hr-ft-°F
   b. Thermal Capacitance, Btu/ft^3-°F

3. Heat Transfer Coefficients

   a. Primary Containment Atmosphere to Primary Containment Wall, Btu/hr-ft^2-°F
   b. Primary Containment Wall to Secondary Containment Atmosphere, Btu/hr-ft^2-°F
   c. Secondary Containment Wall to Secondary Containment Atmosphere, Btu/hr-ft^2-°F
   d. Primary Containment Emissivity, Btu/hr-ft^2-°F
   e. Secondary Containment Emissivity, Btu/hr-ft^2-°F

TABLE 6-18

EVALUATION OF POTENTIAL BYPASS LEAKAGE PATHS
FOR DUAL-CONTAINMENT PLANTS

System	Line Size	Termination Region	Bypass Leakage Barriers	Potential Bypass Path
List all primary containment penetrations by system or line and penetration designation			Key to a list of leakage barriers (e.g., valves, collection systems, closed systems)	(Yes or No)

TABLE 6-19

PRIMARY CONTAINMENT OPERATION
FOLLOWING A DESIGN BASIS ACCIDENT

General

Type of Structure
Appropriate Internal Fission Product Removal Systems
Free Volume of Primary Containment
Mode of Hydrogen Purge (e.g., direct to environs, to recirculation system, to annulus)

Time-Dependent Parameters	Anticipated	Conservative
Leak Rate of Primary Containment		
Leakage Fractions to Volumes Outside the Primary Containment (including the environment).		
Effectiveness of Fission Product Removal Systems		
Initiation of Hydrogen Purge		
Hydrogen Purge Rate		

143 173

TABLE 6-20

SECONDARY CONTAINMENT OPERATION FOLLOWING A DESIGN BASIS ACCIDENT*

General

Type of Structure
Free Volume
Annulus Width (where applicable)
Location of Fission Product Removal Systems

Time-Dependent Parameters	Anticipated	Conservative
Mixing Fraction		
Leak Rate		
Total Recirculation Flow		
Exhaust Flow		
Pressure		
Effectiveness of Fission Product Removal Systems		

*There should be a table such as this for each secondary containment volume.

7. INSTRUMENTATION AND CONTROLS

The reactor instrumentation senses the various reactor parameters and transmits appropriate signals to the regulating systems during normal operation, and to the reactor trip and engineered-safety-feature systems during abnormal and accident conditions. The information provided in this chapter should emphasize those instruments and associated equipment which constitute the protection system (as defined in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"). The analysis of regulating systems and instrumentation should be provided, particularly considerations of regulating system-induced transients which, if not terminated in a timely manner, could result in fuel damage, radiation release, or other public hazard. Details of seismic design and testing should be provided in Section 3.10.

7.1 Introduction

7.1.1 Identification of Safety-Related Systems

List all instrumentation, control, and supporting systems that are safety related, including alarm, communication, and display instrumentation. Distinguish between those systems designed and built by the nuclear steam system supplier and those designed or built by others. Identify the systems that are identical to those of a nuclear power plant of similar design that has recently received a construction permit or an operating license; identify those that are different and discuss the differences and their effects on safety-related systems.

7.1.2 Identification of Safety Criteria

List all design bases (including considerations of instrument errors), criteria, regulatory guides, standards, and other documents that will be implemented in the design of the systems listed in Section 7.1.1.

The specific information identified below should be included in this section of the SAR when it applies equally to all safety-related instrumentation and control systems; otherwise it should be in the section of this chapter that discusses the system to which the information applies.

Provide a description of the technical design bases for all the various functions of the protection system (e.g., scram if reactor vessel water level is ____; this is needed because _____; it is required to operate within ______). In addition to the reactor scram function, bases should be given for all other protection system functions, including engineered safety features, emergency power, interlocks, bypasses, and equipment protection. Diversity requirements should be stated (see IEEE Std 279-1971).

Describe the extent to which the recommendations of the regulatory guides listed below are followed. Wherever alternative approaches are used, demonstrate that an acceptable level of safety has been attained.

143 175

Regulatory Guide 1.11 (Safety Guide 11), "Instrument Lines Penetrating Primary Reactor Containment;"

Regulatory Guide 1.22 (Safety Guide 22), "Periodic Testing of Protection System Actuation Functions;"

Regulatory Guide 1.29, "Seismic Design Classification;"

Regulatory Guide 1.30 (Safety Guide 30), "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment;"

Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants;"

Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems;"

Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems;"

Regulatory Guide 1.62, "Manual Initiation of Protective Actions;"

Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants;"

Regulatory Guide 1.68, "Initial Test Program for Water-Cooled Reactor Power Plants;"

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

Regulatory Guide 1.75, "Physical Independence of Electric Systems." The physical identification of safety-related equipment should also be addressed in this section;

Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems;"

Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants;"

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

Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants;"

Regulatory Guide 1.105, "Instrument Setpoints;" and

143 176

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

7.2 Reactor Trip System

For standardized systems, it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.2.1 Description

7.2.1.1 System Description. Provide a description of the reactor trip system to include initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described. Those parts of any system not required for safety should be identified.

7.2.1.2 Design Basis Information. Provide the design basis information required by Section 3 of IEEE Std 279-1971. Provide preliminary logic diagrams, piping and instrumentation diagrams, and location layout drawings of all reactor trip systems and supporting systems in the PSAR.

7.2.1.3 Final System Drawings. In the FSAR, provide electrical schematic diagrams for all reactor trip systems and supporting systems, final logic diagrams, piping and instrumentation diagrams, and location layout drawings. Describe the differences, if any, between the logic diagrams and schematics submitted in the PSAR and those in the FSAR and the effects on safety-related systems.

7.2.2 Analysis

Provide analyses, including a failure mode and effects analysis, to demonstrate how the requirements of the General Design Criteria, IEEE Std 279-1971, applicable regulatory guides, and other appropriate criteria and standards are satisfied. In addition to postulated accidents and failures, these analyses should include, but not be limited to, considerations of instrumentation installed to prevent or mitigate the consequences of:

1. Spurious control rod withdrawals,
2. Loss of plant instrument air systems,
3. Loss of cooling water to vital equipment,
4. Plant load rejection, and
5. Turbine trip.

The analyses should also discuss the need for and method of changing to more restrictive trip setpoints during abnormal operating conditions such as operation with fewer than all reactor coolant loops operating. Reference may be made to other sections of the SAR for supporting systems.

7.3 Engineered-Safety-Feature Systems

For standardized systems, it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.3.1 Description

7.3.1.1 System Description. Provide a description of the instrumentation and controls associated with the engineered safety features (ESF), including initiating circuits, logic, bypasses, interlocks, sequencing, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described. Those parts of any system not required for safety should be identified.

7.3.1.2 Design Basis Information. Provide the design basis information required by Section 3 of IEEE Std 279-1971. For the PSAR review, provide preliminary electrical schematic diagrams, logic diagrams, piping and instrumentation diagrams, and location layout drawings of all engineered-safety-feature instrumentation, control systems, and supporting systems.

7.3.1.3 Final System Drawings. In the FSAR, provide electrical schematic diagrams for all ESF circuits and supporting systems, and final logic diagrams, piping and instrumentation diagrams, and location layout drawings. Describe the differences, if any, between the logic diagrams and schematics submitted in the PSAR and those in the FSAR and the effects on safety-related systems.

7.3.2 Analysis

Provide analyses, including a failure mode and effects analysis, to demonstrate how the requirements of the General Design Criteria and IEEE Std 279-1971 are satisfied and the extent to which applicable regulatory guides and other appropriate criteria and standards are satisfied. In addition to postulated accidents and failures, these analyses should include considerations of (1) loss of plant instrument air systems and (2) loss of cooling water to vital equipment. The method for periodic testing of engineered-safety-feature instrumentation and control equipment and the effects on system integrity during testing should be described.

7.4 Systems Required for Safe Shutdown

For standardized systems, it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.4.1 Description

Provide a description of the systems that are needed for safe shutdown of the plant, including initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described. Provide the design basis information required by Section 3 of IEEE Std 279-1971. Provide logic diagrams, piping and instrumentation diagrams, and location layout drawings for these systems. In the FSAR, provide electrical schematic diagrams.

Describe the provisions taken in accordance with NRC General Design Criterion 19 to provide the required equipment outside the control room for hot and cold shutdown.

7.4.2 Analysis

Provide analyses that demonstrate how the requirements of the General Design Criteria, IEEE Std 279-1971, applicable regulatory guides, and other appropriate criteria and standards are satisfied. These analyses should include considerations of instrumentation installed to permit a safe shutdown in the event of:

1. Loss of plant instrument air systems,

2. Loss of cooling water to vital equipment,

3. Plant load rejection, and

4. Turbine trip.

7.5 Safety-Related Display Instrumentation

7.5.1 Description

Include a description of the instrumentation systems (including control rod position indicating systems) that provides information to enable the operator to perform required safety functions.

7.5.2 Analysis

Provide an analysis to demonstrate that the operator has sufficient information to perform required manual safety functions (e.g., ensuring safe control rod patterns, manual engineered-safety-feature operations, possible unanticipated postaccident operations, and monitoring the status of safety equipment) and sufficient time to make reasoned judgments and take action where operator action is essential. Identify appropriate safety criteria in the PSAR and demonstrate compliance with these criteria in the FSAR.

143 179

Information should be provided to identify the information readouts or indications provided to the operator for monitoring conditions in the reactor, the reactor coolant system, and in the containment and safety-related process systems, including engineered safety features, throughout all operating conditions of the plant, including anticipated operational occurrences and accident and postaccident conditions (including instrumentation to follow the course of accidents). The information should include the design criteria, the type of readout, number of channels provided, their range, accuracy, and location, and a discussion of the adequacy of the design.

7.6 All Other Instrumentation Systems Required for Safety

This section should contain information on all other instrumentation systems required for safety that are not included under reactor trip, engineered safety features, safe shutdown, safety-related display instrumentation systems, or any of their supporting systems (e.g., fire protection instrumentation and detection systems, cold water slug interlocks, refueling interlocks, and interlocks that prevent overpressurization of low-pressure systems).

7.6.1 Description

Provide a description of all systems required for safety not already discussed, including initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described (reference may be made to other sections of the SAR). Provide the design basis information required by Section 3 of IEEE Std 279-1971. For an FSAR, sufficient schematic diagrams should be provided to permit an independent evaluation of compliance with the safety criteria.

7.6.2 Analysis

Provide analyses to demonstrate how the requirements of the General Design Criteria, IEEE Std 279-1971, applicable regulatory guides, and other appropriate criteria and standards are satisfied. These analyses should include, but not be limited to, considerations of instrumentation installed to prevent or mitigate the consequences of:

1. Cold water slug injections,
2. Refueling accidents,
3. Overpressurization of low-pressure systems, and
4. Fires.

Reference may be made to other sections of the SAR for supporting systems and analyses.

143 180

7.7 Control Systems Not Required for Safety

For standardized systems, it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.7.1 Description

The following information should be provided with regard to the control systems not required for safety:

1. Identification of the major plant control systems (e.g., primary temperature control, primary water level control, steam generator water level control) that are identical to those in a nuclear power plant of similar design by the same nuclear steam system supplier that has recently received a construction permit or an operating license; and

2. A list and discussion of the design differences in those systems not identical to those used in the reference nuclear power plant. This discussion should include an evaluation of the safety significance of each design difference.

7.7.2 Analysis

Provide analyses to demonstrate that these systems are not required for safety. The analyses should demonstrate that the protection systems are capable of coping with all (including gross) failure modes of the control systems.

143 181

8. ELECTRIC POWER

The electric power system is the source of power for the reactor coolant pumps and other auxiliaries during normal operation and for the protection system and engineered safety features during abnormal and accident conditions. The information in this chapter should be directed toward establishing the functional adequacy of the safety-related electric power systems and ensuring that these systems have adequate redundancy, independence, and testability in conformance with current criteria. Details of seismic design and testing should be provided in Section 3.10.

8.1 Introduction

A brief description of the utility grid and its interconnection to other grids should be included, and the onsite electric system should be described briefly in general terms. The safety loads (i.e., the systems and devices that require electric power to perform their safety functions) should be identified; the safety functions performed (e.g., emergency core cooling, containment cooling) and the type of electric power (a.c. or d.c.) required by each safety load should be indicated. The design bases, criteria, regulatory guides, standards, and other documents that will be implemented in the design of the safety-related electric systems should be presented and discussed.

Describe the extent to which the recommendations of the regulatory guides listed below are followed. Wherever alternative approaches are used, demonstrate that an acceptable level of safety has been attained.

Regulatory Guide 1.6 (Safety Guide 6), "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems;"

Regulatory Guide 1.9 (Safety Guide 9), "Selection of Diesel Generator Set Capacity for Standby Power Supplies;"

Regulatory Guide 1.22 (Safety Guide 22), "Periodic Testing of Protection System Actuation Functions;"

Regulatory Guide 1.29, "Seismic Design Classification;"

Regulatory Guide 1.30 (Safety Guide 30), "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment;"

Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants;"

Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants;"

Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments;"

143 182

Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems;"

Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems;"

Regulatory Guide 1.62, "Manual Initiation of Protective Actions;"

Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants;"

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

Regulatory Guide 1.75, "Physical Independence of Electric Systems;"

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

Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants;"

Regulatory Guide 1.93, "Availability of Electric Power Sources;"

Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants;"

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

Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used As Onsite Electric Power Systems at Nuclear Power Plants;"

Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems;"

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

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

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

Indicate whether IEEE Std 387, "Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Stations," is followed. If an alternative approach is used, demonstrate that an acceptable level of safety has been attained.

8.2 Offsite Power System

8.2.1 Description

A system description and an analysis sufficient to demonstrate compliance with 10 CFR Part 50 and the Commission's General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 should be provided. In addition, the SAR should indicate the extent to which the applicant has followed the recommendations of regulatory guides and other applicable standards and criteria (e.g., industry standards normally used by the applicant in the installation of safety systems and internal standards and criteria). In particular, the circuits that supply power for safety loads from the transmission network should be identified and shown to meet GDC 17 and 18. Voltage level and length of each transmission line from the site to the first major substation that connects the line to the grid should be provided. All unusual features of these transmission lines should be described (e.g., crossovers or proximity of other lines, rugged terrain, vibration or galloping conductor problems, icing or other heavy loading conditions, and high thunderstorm occurrence rate). Describe and provide layout drawings of the circuits that connect the onsite distribution system to the preferred power supply; include transmission lines, switchyard arrangement, rights-of-way, etc.

8.2.2 Analysis

The results of steady-state and transient stability analyses should be provided to demonstrate compliance with the final paragraph of GDC 17. In determining the most critical transmission line, consider lines that use a common tower to be a single line. Provide information and a discussion of grid availability, including the frequency, duration, and causes of outages.

8.3 Onsite Power Systems

8.3.1 A.C. Power Systems

8.3.1.1 Description. Describe the onsite a.c. power systems with emphasis placed on those portions of the systems that are safety related. Those portions that are not related to safety need only be described in sufficient detail to permit an understanding of their interactions with the safety-related portions. The description of the safety-related portions should include:

1. Power supply feeders (i.e., network configuration),

2. Busing arrangements,

3. Loads supplied from each bus,

4. Manual and automatic interconnections between buses, buses and loads, and buses and supplies,

5. Interconnections between safety-related and non-safety-related buses,

6. Redundant bus separation,

7. Equipment capacities,

8. Automatic loading and stripping of buses,

9. Safety-related equipment identification,

10. Instrumentation and control systems for the applicable power systems with the assigned power supply identified,

11. Electric circuit protection system network (e.g., selective trip), including setting criteria,

12. The scheme for testing these systems during power operation, and

13. Any systems and equipment shared between units.

The basis for the power required for each safety load (e.g., motor nameplate rating, pump runout condition, or estimated load under expected flow and pressure) should be given. The continuous and short-term ratings for the onsite power source should be provided. In some cases, the basis for the requested information is engineering judgment or correlation with other similar plants; nevertheless, the information requested should be submitted and all limitations cited. The FSAR should completely update all previously transmitted information and should verify that all systems are adequately sized and that all pertinent criteria are met.

The following design aspects of the onsite emergency electric power sources (e.g., diesel generators) should be described in preliminary form in the PSAR:

1. Starting initiating circuits,

2. Starting mechanism and system,

3. Tripping devices,

4. Interlocks,

5. Permissives,

6. Load shedding circuits,

7. Testability,

8. Fuel oil storage and transfer system,

9. Cooling and heating systems,

10. Instrumentation and control systems, including status alarms and indications, with assigned power supply, and

11. Prototype qualification program.

This description should be complete in the FSAR. Any features or components not previously used in similar applications in nuclear generating stations should be identified. Provide single-line diagrams of the onsite a.c. distribution systems, including identification of all safety loads. The physical arrangement of the components of the system should be described in sufficient detail to permit independent verification that single events and accidents will not disable redundant features. Sufficient plant layout drawings should be provided to permit evaluation of the physical separation and isolation of redundant portions of the system. The PSAR should provide a table that illustrates the automatic and manual loading and unloading of each standby power supply. The FSAR should provide an updated table reflecting any changes or revisions. Include the time (sequence) of each event, size of load, inrush current or starting kVA, identification of redundant equipment, and length of time each load is required. For the safety-related systems, describe the bases and provide the design criteria that establish:

1. Motor size,

2. Minimum motor accelerating voltage,

3. Motor starting torque,

4. Minimum motor torque margin over pump torque through accelerating period,

5. Motor insulation,

6. Temperature monitoring devices provided in large horsepower motors,

7. Interrupting capacity of switchgear, load centers, control centers, and distribution panels,

8. Electric circuit protection, and

9. Grounding requirements.

The FSAR should identify all deviations from these criteria as described in the PSAR and provide justification for any deviations. Sufficient logic and schematic diagrams should be provided to permit an independent evaluation of compliance with the safety criteria.

143 186

8.3.1.2 Analysis. Provide analyses to demonstrate compliance with the Commission's General Design Criteria and to indicate the extent to which the recommendations of regulatory guides and other applicable criteria are followed. Especially important are the analyses to demonstrate compliance with GDC 17 and 18 and the discussion to indicate the extent to which the recommendations of Regulatory Guides 1.6 and 1.9 (Safety Guides 6 and 9) and of Regulatory Guide 1.32 are followed. The discussion should identify all aspects of the onsite power system that do not conform to Regulatory Guides 1.6, 1.9, and 1.32 and should explain why such deviations are not in conflict with applicable General Design Criteria.

Identify all safety-related equipment that must operate in a hostile environment (e.g., radiation, temperature, pressure, humidity) during and/or subsequent to a postulated accident (e.g., loss-of-coolant accident, steam line break). All the conditions under which the equipment must operate should be tabulated. Provide bases, criteria, and analyses of the potential effects of (1) radiation (i.e., radiation due to accident conditions superimposed on that for long-term normal operation) on safety-related electric equipment throughout the plant and (2) loss-of-coolant accidents or steam line breaks on all safety-related electric equipment within primary reactor containment (e.g., motors, cables) that must operate during and/or subsequent to such an accident. The successful completion of any applicable qualification tests for the above cases should be documented. Where such tests have not been previously completed, plans and schedules of the qualification tests proposed should be documented. The FSAR should document the results of these tests.

8.3.1.3 Physical Identification of Safety-Related Equipment. Describe the means proposed to identify physically the onsite power system equipment as safety-related equipment in the plant to ensure appropriate treatment, particularly during maintenance and testing operations. The description should include the method used to readily (without the necessity for consulting reference material) distinguish between redundant Class 1E systems, associated circuits assigned to redundant Class 1E divisions, and non-Class 1E systems.

8.3.1.4 Independence of Redundant Systems. Present the criteria and their bases that establish the minimum requirements for preserving the independence of redundant Class 1E electric systems* through physical arrangement and separation and for ensuring the minimum required equipment availability during any design basis event.* A discussion should be included of the administrative responsibility and control to be provided to ensure compliance with these criteria during the design and installation of these systems. The criteria and bases for the installation of electrical cable for these systems should, as a minimum, include a description of the extent to which the recommendations of Regulatory Guide 1.75, "Physical Independence of Electric Systems," are followed.

*Class 1E electric systems and design basis events are defined in IEEE Std 308-1971.

8.3.2 D.C. Power Systems

8.3.2.1 Description. A description of the d.c. power systems clearly delineating the safety-related portions should be provided. The non-safety-related portion need only be described in sufficient detail to permit an understanding of its interaction with the safety-related portions. The description of the safety-related portion should include requirements for separation, capacity, charging, ventilation, loading, redundancy, and testing. The safety loads should be clearly identified, and the length of time they would be operable in the event of loss of all a.c. power should be stated. Sufficient schematic diagrams should be provided in the FSAR to permit an independent evaluation of compliance with the safety criteria.

8.3.2.2 Analysis. Provide an analysis to demonstrate compliance with the Commission's General Design Criteria, and describe the extent to which recommendations of regulatory guides and other applicable criteria are followed. The same information described in Sections 8.3.1.2 and 8.3.1.3, as applicable, should be provided.

8.3.3 Fire Protection for Cable Systems

The measures employed for the prevention of and protection against fires in electrical cables should be described in Section 9.5.1. The following should be described in Sections 8.3.1 and 8.3.2:

1. Cable derating and cable tray fill, and

2. Fire barriers and separation between redundant trays.

143 188

9. AUXILIARY SYSTEMS

This chapter should provide information concerning the auxiliary systems included in this facility. The information in the PSAR should reflect the preliminary design of the auxiliary systems, and the FSAR information should reflect the final design.

Those systems that are essential for the safe shutdown of the plant or the protection of the health and safety of the public should be identified. The description of each system, the design bases for the system and for critical components, a safety evaluation demonstrating how the system satisfies the design bases, the testing and inspection to be performed to verify system capability and reliability, and the required instrumentation and controls should be provided. There may be aspects of the auxiliary systems that have little or no relationship to protection of the public against exposure to radiation. In such cases, enough information should be provided to allow understanding of the auxiliary system design and function with emphasis on those aspects of design and operation that might affect the reactor and its safety features or contribute to the control of radioactivity.

The capability of the system to function without compromising the safe operation of the plant under both normal operating or transient situations should be clearly shown by the information provided, i.e., a failure analysis.

Seismic design classifications should be stated with reference to detailed information provided in Chapter 3, where appropriate. Radiological considerations associated with operation of each system under normal and accident conditions, where applicable, should be summarized and reference made to detailed information in Chapters 11 or 12 as appropriate.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Design Bases. The design bases for new fuel storage facilities should be provided and should include such considerations as quantity of fuel to be stored, means for maintaining a subcritical array, and the degree of subcriticality provided for the most reactive condition possible together with the assumptions used in this calculation and design loadings to be withstood.

9.1.1.2 Facilities Description. A description of the new fuel storage facilities, including drawings, and location in the station complex should be provided.

9.1.1.3 Safety Evaluation. An evaluation of the capability of the new fuel storage facilities to reduce the probability of occurrence of unsafe conditions should be presented and should include the degree of

143 189

subcriticality, governing codes for design, ability to withstand external loads and forces, and safety implications related to sharing (for multi-unit facilities). Details of the seismic design and testing should be presented in Section 3.7.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases. The design bases for the spent fuel storage facilities should be provided and should include such considerations as quantity of fuel to be stored, means for maintaining a subcritical array, degree of subcriticality provided together with the assumptions used in this calculation, shielding requirements, and design loadings to be withstood.

9.1.2.2 Facilities Description. A description of the spent fuel storage facilities, including drawings, and location in the station complex should be provided.

9.1.2.3 Safety Evaluation. An evaluation of the protection of the spent fuel storage facilities against unsafe conditions should be presented and should include the degree of subcriticality, governing codes for design, ability to withstand external loads and forces, ability to ensure continuous cooling, provisions to avoid accidental dropping of heavy objects on spent fuel, material compatibility requirements, radiological considerations (details should be presented in Chapter 12), ability of the fuel storage racks to withstand lifting forces if a fuel assembly accidentally engages a rack while being lifted, and safety implications related to sharing (for multi-unit facilities). Additional guidance regarding acceptable design of the spent fuel storage facilities is given in Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

9.1.3 Spent Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases. The design bases for the cooling and cleanup system for the spent fuel facilities should be provided and should include the requirements for continuous or intermittent cooling, the quantity of spent fuel to be cooled, the requirements for pool water temperature and cleanliness from fission and corrosion products, makeup requirements, and level and radiation shielding requirements.

9.1.3.2 System Description. A description of the cooling and cleanup system, including a description of the instrumentation utilized, should be provided. The FSAR should include a detailed updated description and drawings.

9.1.3.3 Safety Evaluation. An evaluation of the cooling system, including the capability for spent fuel cooling during normal and abnormal conditions, provisions to ensure that pool water will not be lost at a rate greater than the makeup capability, and ability to maintain acceptable pool water conditions, should be provided. The radiological evaluation

143 190

of the cleanup system should be presented in Chapters 11 and 12. Additional guidance regarding acceptable coolant makeup requirements is given in Regulatory Guide 1.13.

9.1.3.4 Inspection and Testing Requirements. The inspection and testing requirements for the cooling and cleanup system should be described.

9.1.4 Fuel Handling System

9.1.4.1 Design Bases. The design bases for the fuel handling system (FHS) should be provided; the performance and load handling requirements, handling control features, and provisions to prevent fuel handling and cask drop accidents should be included.

9.1.4.2 System Description. A description of the FHS, including all components for transporting and handling fuel from the time it reaches the plant until it leaves the plant, should be provided. Descriptions of the containment polar crane and spent fuel cask handling crane should be included. An outline for the procedures used in new fuel receipt and storage, reactor refueling operations, and spent fuel storage and shipment should be provided. Component drawings, building layouts, and illustrations of the fuel handling procedures should also be provided. Detailed descriptions and drawings should be included in the FSAR. Design data, seismic category, and quality class should be provided for all principal components. The design codes and standards used for design, manufacture, testing, maintenance and operation, and seismic design aspects should be enumerated.

9.1.4.3 Safety Evaluation. The safety evaluation should demonstrate that the system design meets the applicable redundancy and diversity requirements. It should be demonstrated that the FHS design precludes inadvertent operations or equipment malfunctions or failures that could prevent safe shutdown of the reactor or cause a release of radioactivity. The results of a failure mode and effects analysis should be presented to demonstrate that the individual subsystems and components, including controls and interlocks, are designed to meet the single-failure criterion without compromising the capability of the system to perform its safety functions.

Compliance of the system with applicable General Design Criteria should be demonstrated. The extent to which the recommendations of applicable regulatory guides are followed should be indicated. It should be shown that the seismic design of the individual components will preclude system malfunctions that could prevent safe shutdown of the reactor or cause a release of radioactivity in the event of the Safe Shutdown Earthquake (SSE). It should also be shown that the component design standards and safety factors are adequate.

It should be demonstrated that failure of any part of the spent fuel cask handling crane will not cause any damage to spent fuel and safety-related equipment. This could be accomplished by using a crane design that will prohibit cask drop in the event of a single failure or by adequate facility design that prevents damaging the spent fuel and safety-related equipment for any manner of cask drop, including drop of a tilted cask.

9.1.4.4 Inspection and Testing Requirements. The inspection and testing requirements for the FHS subsystems and components should be described, including shop tests, preoperational tests, and periodic operational tests.

9.1.4.5 Instrumentation Requirements. The system instrumentation and controls should be described. The adequacy of safety-related interlocks to meet the single-failure criterion should be demonstrated.

9.2 Water Systems

This section of the SAR should provide discussions of each of the water systems associated with the plant. Because these auxiliary water systems vary in number, type, and nomenclature for various plant designs, the Standard Format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.2.1 through 9.2.X) for each of the systems. As they apply to a particular plant, these subsections should provide the following information:

1. Design bases,

2. System description, including drawings,

3. Safety evaluation,

4. Testing and inspection requirements, and

5. Instrumentation requirements for each system.

The following paragraphs provide examples of systems that should be discussed, as appropriate to the individual plant, and identify some specific information that should be provided in addition to the items identified above. The examples are not intended to be a complete list of systems to be discussed in this section.

9.2.1 Station Service Water System

Describe the capability of the service water system to meet the single-failure criterion (when this system is safety related), the ability to withstand adverse environmental occurrences, requirements for normal operation and for operating during and subsequent to postulated accident conditions, including loss of offsite power, provisions for reactor compartment flooding during the post-LOCA period, if required, and the ability of the system to detect and prevent excessive leakage of radioactive material to the environment. Include a failure analysis to demonstrate that a single failure will not result in the loss of all or an unacceptable portion of the cooling function (considering failures of active and passive components and diverse sources of electric power for pumps, valves, and control purposes), capability of the system to function during abnormally

143 192

high and low water levels, prevention of long-term corrosion and organic fouling that may degrade system performance, and safety implications related to sharing (for multi-unit facilities). Reference Section 3.6 with respect to the analysis of postulated cracks in moderate-energy piping systems and Sections 2.4.11.5, 2.4.11.6, and 2.4.12 where applicable.

9.2.2 Cooling System for Reactor Auxiliaries

Discuss the capability of the reactor system auxiliaries to meet the single-failure criterion when required, the ability to withstand adverse environmental occurrences, requirements for normal operation and for operating during and subsequent to postulated accident conditions, including loss of offsite power, and requirements for leakage detection and containment of leakage. Include a failure analysis to demonstrate that a single failure will not result in the loss of all, or an unacceptable portion of, the cooling function (considering failures of active and passive components, and diverse sources of electric power for pumps, valves, and control purposes), the means for preventing or controlling leakage of activity to the outside environment, leakage detection provisions, prevention of long-term corrosion that may degrade system performance, and safety implications related to sharing (for multi-unit facilities). Reference Section 3.6 with respect to the analysis of postulated cracks in moderate-energy piping systems.

9.2.3 Demineralized Water Makeup System

9.2.4 Potable and Sanitary Water Systems

A description of the potable and sanitary water systems should be provided. System design criteria should provide for prevention of connections to systems having the potential for containing radioactive material. An evaluation of radiological contamination, including accidental, and safety implications of sharing (for multi-unit facilities) should be described.

9.2.5 Ultimate Heat Sink

A description of the ultimate heat sink to be used to dissipate waste heat from the plant during normal, shutdown, and accident conditions should be provided. Additional guidance regarding acceptable features of the ultimate heat sink is given in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants." Reference Sections 2.3.1.2 and 2.4.11 where applicable.

9.2.6 Condensate Storage Facilities

A discussion of the environmental design considerations, requirements for leakage control (including mitigation of environmental effects), limits for radioactivity concentration, code design requirements, and material compatibility and corrosion control should be given. An analysis of storage facility failure and provisions for mitigating environmental effects should be provided. The evaluation of radiological considerations should be presented in Chapter 12.

9.3 Process Auxiliaries

This section of the SAR should provide discussions of each of the auxiliary systems associated with the reactor process system. Because these auxiliary systems vary in number, type, and nomenclature for various plant designs, the Standard Format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.3.1 through 9.3.X) for each of the systems. These subsections should provide the following information:

1. Design bases,

2. System description,

3. Safety evaluation,

4. Testing and inspection requirements, and

5. Instrumentation requirements for each system.

The following paragraphs provide examples of systems that should be discussed, as appropriate to the individual plant, and identify some specific information that should be provided in addition to the items identified above. The examples are not intended to be a complete list of systems to be discussed in this section. For example, the boron recovery system and the failed fuel detection system should both be discussed in this section.

9.3.1 Compressed Air Systems

Describe the compressed air systems that provide station air for service and maintenance uses, and include discussion of provisions for meeting the single-failure criterion for safety-related compressed air systems, air cleanliness and quality requirements, and environmental design requirements. The evaluation of the compressed air system should include a failure analysis (including diverse sources of electric power), maintenance of air cleanliness to ensure system reliability, the capability to isolate if required, and safety implications related to sharing (for multi-unit plants).

9.3.2 Process Sampling System

Describe the sampling system for the various plant fluids. The design bases should include consideration of sample size and handling to ensure that a representative sample is obtained; provisions for isolation of the system and the means to limit reactor coolant losses; requirements to minimize, to the extent practical, hazards to plant personnel; and system pressure, temperature, and code requirements. The points from which samples will be obtained should be delineated. The evaluation of the sampling system should provide assurance that representative samples will be obtained and that sharing (for multi-unit facilities) will not adversely affect plant safety.

9.3.3 Equipment and Floor Drainage System

Describe the drainage systems for collecting the effluent from high activity and low activity liquid drains from various specified equipment items and buildings. Design considerations for precluding backflooding of equipment in safety-related compartments should be discussed. Areas where the drainage system is used to detect leakage from safety systems should be identified. Design considerations for preventing transfer of contaminated fluids to noncontaminated drainage systems should be discussed. An evaluation of radiological considerations for normal operation and postulated spills and accidents, including the effects of sharing (for multi-unit plants), should be presented in Chapters 11 and 12.

9.3.4 Chemical and Volume Control System (PWRs) (Including Boron Recovery System)

9.3.4.1 Design Bases. The design bases for the chemical and volume control system (CVCS) and the boron recovery system (BRS) should include consideration of (1) the capability to vary coolant chemistry for control of reactivity and corrosion and (2) the capability for maintaining the required reactor coolant system inventory and the reactor coolant pump seal water requirements. Items to be considered include the maximum and normal letdown flow rates, charging rates for both normal operation and maximum leakage conditions, boric acid storage requirements for reactivity control, water chemistry requirements, and boric acid and primary water storage requirements in terms of maximum number of startup and shutdown cycles.

9.3.4.2 System Description. A complete description of the system and components, including piping and instrumentation diagrams, should be provided. Design data, seismic category, and quality class should be provided for all components. The principles of system operation, both automatic and manual, should be provided for steady-state, transient, startup, shutdown, and accident conditions. A discussion on reactor coolant water chemistry requirements should be provided. Temperature control provisions for line heat tracing and tank heating, including provision for alarm failures, should be described. Tabulations of system design parameters and component design data should be provided.

9.3.4.3 Safety Evaluation. The safety evaluation should demonstrate that the system is designed to provide for safe operation and shutdown and to prevent or mitigate certain postulated accidents. This includes demonstration that the system boron inventory is adequate for the most stringent cold shutdown requirements, including anticipated operational occurrences. Provisions to prevent loss of solubility of boric acid solutions should also be discussed. This section should also include

143 195

demonstration that the system has the pumping capability to supply reactor coolant makeup for protection against small pipe or component failures. The safety evaluation should demonstrate that the system is designed to limit radioactive releases to the environment to allowable limits for both normal operation and accident conditions. The adequacy of the component and piping seismic design category and quality class should be justified. The results of a failure mode and effects analysis should be presented to demonstrate that the system can meet the single-failure criterion without compromising safe plant shutdown and the ability to prevent or mitigate postulated accidents.

Compliance of the system with applicable General Design Criteria should be demonstrated. The extent to which the recommendations of applicable regulatory guides are followed should be indicated. It should be shown that the essential portions of the system will be protected from failure of non-Seismic Category I equipment and piping and also from flooding, tornadoes, internally and externally generated missiles, and the effects of high- and moderate-energy line failures.

9.3.4.4 Inspection and Testing Requirements. The inspection and testing requirements for the CVCS should be described.

9.3.4.5 Instrumentation Requirements. The system instrumentation and controls should be described. The adequacy of safety-related instrumentation and controls to fulfill their functions should be demonstrated.

9.3.5 Standby Liquid Control System (BWRs)

9.3.5.1 Design Bases. The design bases for the standby liquid control system (SLCS) should include consideration of the capability for reactor shutdown independent of the normal reactivity control system with a reasonable shutdown margin at any time in core life, system redundancy, and ability to periodically verify functional performance capability.

9.3.5.2 System Description. A description of the system and components, complete with piping and instrumentation diagrams, should be provided. Temperature control provisions for line heat tracing and tank heating, including provisions for alarm failures, should be described. Design data, seismic category, and quality class should be provided for all components. The principles of system operation and testing should be provided.

9.3.5.3 Safety Evaluation. The safety evaluation should demonstrate that the system has adequate storage capacity and injection rate to bring the reactor from rated power to cold shutdown at any time in core life (control rods withdrawn in the rated power pattern) with adequate margin for adverse factors, including xenon decay, elimination of steam voids, allowance for imperfect mixing, leakage, and dilution. Provisions to prevent loss of solubility of sodium pentaborate solutions should be discussed. The adequacy of the component and piping seismic design category should be justified.

143 196

The results of a failure mode and effects analysis should be presented to demonstrate that the system can meet the single-failure criterion without compromising the shutdown capability of the system. Compliance of the system with applicable General Design Criteria should be demonstrated. The extent to which the recommendations of applicable regulatory guides are followed should be indicated. It should be shown that the essential portions of the system will be protected from failure of non-seismic equipment and piping and also from flooding, tornadoes, internally and externally generated missiles, and the effects of high- and moderate-energy line failures.

9.3.5.4 Inspection and Testing Requirements. The inspection and testing requirements for the SLCS, including periodic operational testing, should be described.

9.3.5.5 Instrumentation Requirements. The system instrumentation and controls should be described. The adequacy of safety-related instrumentation and controls to fulfill their functions should be demonstrated.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

Following are examples of systems that should be discussed, as appropriate to the individual plant. Some specific information that should be provided is also identified. The examples are not intended to be a complete list of systems to be discussed in this section. For example, the ventilation system for both the diesel building and the containment ventilation system should be described in this section.

9.4.1 Control Room Area Ventilation System

9.4.1.1 Design Bases. The design bases for the air treatment system for the control room and other auxiliary rooms (e.g., relay rooms and emergency switchgear rooms) considered to be part of the control areas should be provided. Include the design criteria (e.g., single failure), requirements for the manual or automatic actuation of system components or isolation dampers, ambient temperature and humidity requirements, criteria for plant operator comfort and safety, requirements for radiation protection and monitoring of abnormal radiation levels and other airborne contaminants, and environmental design requirements.

9.4.1.2 System Description. A description, including preliminary piping and instrumentation diagrams, of the air treatment systems for the control room should be presented in the PSAR. A detailed updated description and piping and instrumentation diagrams should be provided in the FSAR.

9.4.1.3 Safety Evaluation. A safety evaluation of the control room air treatment system should be provided. The evaluation should include the following subjects. (If these subjects are dealt with elsewhere in the SAR, a summary discussion should be presented here and the sections that include the details should be referenced.)

143 197

1. Detection of adverse or dangerous environmental conditions (smoke, radiation, etc.),

2. Capability to exclude entry of contaminants (zone pressurization and isolation),

3. Capability for the removal of contamination by filtration (also see Section 6.5.1, ESF Filters),

4. Removal of contamination by purging, and

5. Maintenance of acceptable zone temperature and humidity and anticipated degradation of equipment performance if temperature limits are exceeded.

Additional detailed discussion of control room ventilation systems should appear in Section 6.4, "Habitability Systems," and in paragraph 5, "Radiological consequences," of Section 15.X.X.

9.4.1.4 Inspection and Testing Requirements. The inspection and testing requirements for the control room air treatment system should be described.

9.4.2 Spent Fuel Pool Area Ventilation System

9.4.2.1 Design Bases. The design bases of the ventilation system for the spent fuel pool area should be provided. Include the requirements for meeting the single-failure criterion, seismic design criteria, requirements for the manual or automatic actuation of system components or isolation dampers, ambient temperature limits, preferred direction of airflow from areas of low potential radioactivity to areas of high potential radioactivity, monitoring normal and abnormal radiation levels within the area, differential pressures to be maintained and measured, and the requirements for the treatment of exhaust air. Details of the means for protection of system vents or louvers from missiles should be provided.

9.4.2.2 System Description. A description, including preliminary piping and instrumentation diagrams, of the spent fuel pool area ventilation system should be presented in the PSAR. In the FSAR, provide a detailed description and piping and instrumentation diagrams.

9.4.2.3 Safety Evaluation. An evaluation of the spent fuel area ventilation system and results from failure mode and effects analysis should be provided. Include a discussion of the ability to (1) detect radiation in the area of the spent fuel pool and (2) filter the contaminants out of the air before exhausting it to the environment or prevent the contaminated air from leaving the spent fuel area.

9.4.2.4 Inspection and Testing Requirements. The inspection and testing requirements for the spent fuel area ventilation system should be described.

143 198

9.4.3 Auxiliary and Radwaste Area Ventilation System

9.4.3.1 Design Bases. The design bases for the air handling system for the radwaste area and the areas of the auxiliary building containing safety-related equipment should be presented. Include requirements for meeting the single-failure criterion, seismic design criteria, requirements for the manual or automatic actuation of system components or isolation dampers, ambient temperature limits, preferred direction of airflow from areas of low potential radioactivity to areas of high potential radioactivity, differential pressures to be maintained and measured, requirements for the monitoring of normal and abnormal radiation levels, and requirements for the treatment of exhaust air. Details of the means for protection of system vents or louvers from missiles should be provided.

9.4.3.2 System Description. A description, including preliminary piping and instrumentation diagrams, of the air handling system for the auxiliary and radwaste area should be presented in the PSAR. Detailed updated piping and instrumentation diagrams should be provided in the FSAR.

9.4.3.3 Safety Evaluation. An evaluation of the auxiliary and radwaste area ventilation system should be presented and should include a system failure analysis (including the effects of inability to maintain preferred airflow patterns). Evaluation of radiological consideration for normal operation should be presented in Chapters 11 and 12.

9.4.3.4 Inspection and Testing Requirements. The inspection and testing requirements for the auxiliary and radwaste area ventilation system should be described.

9.4.4 Turbine Building Area Ventilation System

9.4.4.1 Design Bases. The design bases for the air handling system for the turbine-generator area in the turbine building should be presented. Include requirements for the manual or automatic actuation of system components or isolation dampers, ambient temperature limits, preferred direction of airflow from areas of low potential radioactivity to areas of higher potential radioactivity, requirements for monitoring of abnormal radiation levels, and requirements for treatment of exhaust air.

9.4.4.2 System Description. A description, including preliminary piping and instrumentation diagrams, of the air handling system for the turbine building should be provided in the PSAR. A detailed updated description and piping and instrumentation diagrams should be provided in the FSAR.

9.4.4.3 Safety Evaluation. An evaluation of the turbine building air handling system should be presented and should include a system failure analysis (including effects of inability to maintain preferred airflow patterns). Radiological considerations for normal operation should be evaluated in Chapters 11 and 12.

 143 199

9.4.4.4 Inspection and Testing Requirements. The inspection and testing requirements for the turbine building air handling system should be described.

9.4.5 Engineered-Safety-Feature Ventilation System

9.4.5.1 Design Bases. The design bases for the air handling system for the areas housing engineered-safety-feature equipment should be presented. Include requirements for meeting the single-failure criterion, requirements for the manual or automatic actuation of system components or isolation dampers, ambient temperature requirements, preferred direction of airflow from areas of low potential radioactivity to areas of higher potential radioactivity, and the requirements for the monitoring of normal and abnormal radiation levels. Details of the means for protection of system vents or louvers from missiles should be provided.

9.4.5.2 Systems Description. A description, including preliminary piping and instrumentation diagrams, of the air handling system for the engineered-safety-feature area should be presented in the PSAR. A detailed updated description and piping and instrumentation diagrams should be provided in the FSAR.

9.4.5.3 Safety Evaluation. An evaluation of the engineered-safety-feature ventilation system should be presented and should include a system failure analysis. An analysis should be provided to demonstrate that a component necessary for safe shutdown or to mitigate the consequences of an accident can perform its safety function when subjected to ambient temperatures and conditions associated with the loss of the engineered-safety-feature ventilation system during an accident condition coincident with the loss of offsite power. The effect of redundant systems may be included in the evaluation.

9.4.5.4 Inspection and Testing Requirements. The inspection and testing requirements for the engineered-safety-feature ventilation system should be provided.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

9.5.1.1 Design Bases.

1. The PSAR should identify those areas where a fire could affect, either directly or indirectly, Seismic Category I safety-related structures, systems, or components.

2. The concept of defense in depth (using echelons of safety systems to achieve a required high level of safety) is used in the design and operation of nuclear power plants. When applied to the fire protection program, the defense-in-depth principle results in a balance in:

143 200

a. Preventing fires from starting;

b. Detecting fires quickly, suppressing those fires that occur, putting them out quickly, and limiting their damage; and

c. Designing plant safety systems so that a fire that starts in spite of the fire prevention program and burns for a considerable time in spite of fire protection activities will not prevent essential plant safety functions from being performed.

Although no one of these echelons can be perfect or complete by itself, strengthening any one can compensate in some measure for weaknesses, known or unknown, in the others.

3. The primary objective of the fire protection program is to minimize both the probability and consequences of postulated fires. In spite of steps taken to reduce the probability of fire, fires are expected to occur. Therefore, means are needed to detect and suppress fires with particular emphasis on providing passive and active fire protection of appropriate capability and adequate capacity for the systems necessary to achieve and maintain safe plant shutdown with or without offsite power. For other safety-related systems, the fire protection should ensure that a fire will not cause the loss of function of such systems, even though loss of redundancy within a system may occur as a result of the fire.

Generally, in plant areas where the potential fire damage may jeopardize safe plant shutdown, the primary means of fire protection should consist of fire barriers and fixed automatic fire detection and suppression systems. However, total reliance should not be placed on a single fire suppression system. Appropriate backup fire suppression capability should also be provided throughout the plant to limit the extent of fire damage. Portable equipment consisting of hoses, nozzles, portable extinguishers, complete personnel protective equipment, and air breathing equipment should be provided for use by properly trained firefighting personnel. Access for effective manual application of fire extinguishing agents to combustibles should be provided. The adequacy of fire protection for any particular plant safety system or area should be determined by analysis of the effects of the postulated fire relative to maintaining the ability to safely shut down the plant and minimize radioactive releases to the environment in the event of a fire.

4. Fire protection should start with design and must be carried through all phases of construction and operation. A quality assurance (QA) program is needed to identify and rectify errors in design, construction, and operation and is an essential part of defense in depth.

5. The consequences of inadvertent operation of, or a crack in, a moderate-energy line in the fire suppression system should meet the guidelines specified for moderate-energy systems outside containment.

143 201

6. The FSAR should list any unusually hazardous materials to be used on the site that could present unexpected fire hazards or complicate firefighting activities. Such a material listing would include but not be limited to:

a. Flammable liquids,

b. Strong oxidizing agents,

c. Compressed gases, both flammable and nonflammable,

d. Corrosive materials, both acids and caustics, and

e. Explosives or highly flammable materials.

The listing should indicate the amounts of each material to be used, where in the plant and under what conditions each is to be used, and the expected time duration of use.

9.5.1.2 Systems Description

1. The SAR should discuss and list the features of building and facility arrangements and the structural design features that contribute to fire prevention and fire control. List and describe in the discussion the means of egress, fire barriers, and isolation and containment features provided for flame, heat, hot gases, smoke, and other contaminants. Fire barriers with a minimum fire resistance rating of 3 hours should be used. Interior wall and structural components, thermal insulation materials, radiation shielding materials, and soundproofing should be noncombustible.

The SAR should include drawings and a list of equipment and devices that adequately define the principal and auxiliary fire protection systems.

2. The SAR should state the basic requirements used for the design of the fire water supply and distribution systems. It should also specify any particular seismic requirements imposed on the design of each type of fire protection system used in the plant.

3. The SAR should list the various codes and standards used for design and installation of the plant fire protection system.

4. The SAR should discuss (for multi-unit sites) the special fire hazards created and the protection required for an operating unit during the construction of additional units.

5. The SAR should provide a general description of each fire protection system. It should include preliminary drawings that outline each item of fire protection equipment and the complete fire protection system, showing each in relation to safety-related structures, systems, and components for the entire plant.

6. The SAR should discuss the protection and extinguishing systems provided to protect the control room and other operating areas containing safety-related equipment, Class 1E equipment, and cables.

7. The SAR should describe the design features of detection systems, alarm systems, automatic fire suppression systems, and manual, chemical, and gas systems for fire detection, confinement, control, and extinguishment. Discuss the relationship of the fire protection systems (detection and suppression) to the onsite a.c. and d.c. power sources (emergency power supplies).

8. The SAR should discuss smoke, heat, and flame control; combustible and explosive gas control; and toxic contaminant control, including the operating functions of the ventilating and exhaust systems during the period of fire extinguishing and control.

9. The SAR should discuss the fire annunciator warning system, the alarm detection system in the proposed fire protection systems, and the backup or public fire protection suppression capabilities to be provided.

10. The SAR should describe electrical cable fire protection and detection and the fire containment, control, and extinguishing systems provided. Define integrity of the essential electric circuitry needed during the fire for safe shutdown of the plant and for firefighting. Describe the provisions made for protecting this essential electrical circuitry from the effects of fire-suppressing agents.

Cable and cable tray penetration of fire barriers (vertical and horizontal) should be sealed to give protection at least equivalent to that required of the fire barrier.

Electric cable constructions should, as a minimum, pass the flame test in the current IEEE Std 383, "IEEE Standard of Type Test of Class 1E Electrical Cables, Field Splices and Connections for Nuclea Power Generating Stations." (This does not imply that cables passing this test will not require fire protection.)

9.5.1.3 Safety Evaluation (Fire Hazards Analysis)

1. The overall fire protection program should allow the plant to maintain the ability to perform safe shutdown functions and minimize radioactive releases to the environment in the event of a fire. A major element of this program should be the evaluation of potential fire hazards throughout the plant and the effect of postulated fires on safety-related plant areas.

2. The fire hazards for each area identified in paragraph 1 of Section 9.5.1.1 should be evaluated in the SAR. This evaluation should consider (a) fuel loading, considering both fixed and transient combustibles, (b) the expected rate of fire development and maximum intensity,

as these relate to fire detection response sensitivity, and automatic and manual firefighting activities, and (c) generation of smoke and other combustion products, considering both toxic and corrosive characteristics. Show for each of the postulated events that the combination of fire barriers and accessibility make effective manual firefighting feasible.

3. The SAR should postulate initiation of fire in each area at the location that will produce the most severe fire. An ignition source is to be assumed present. Fire development should consider the potential for involvement of other combustibles, both fixed and transient, in the fire area. Where automatic suppression systems are installed, the effects of the postulated fire should be evaluated with and without actuation of such systems.

4. The SAR should provide a failure mode and effects analysis that demonstrates that operation of the fire protection system in areas containing engineered safety features would not produce an unsafe condition or preclude safe shutdown. The effects of firefighting activities and fire suppression agents on safety systems should be discussed. An evaluation of the effects of failure of any portion of the fire protection system not designed to Seismic Category I requirements should be provided with regard to the possibility of damaging other Seismic Category I equipment. An analysis of the fire detection and protection system with regard to design features to withstand the effects of single failures should be included.

5. The SAR should evaluate the effects of postulated fires on safety-related structures, systems, or components for each area of the plant identified above. The discussion should cover the use of noncombustible and fire-resistant materials.

6. The SAR should provide:

a. A complete set of drawings, including, but not limited to, pertinent details of construction, location of rooms and areas, location of fire detection and suppression systems, and fire water mains and hydrants.

b. A listing by fire area of mechanical and electrical equipment both safety and nonsafety related.

c. A listing by fire area of permanent and reasonably expected transient combustibles.

d. A listing by area of fire detection systems showing type of installation and basis for type selected.

e. A listing by area of both primary and backup fire suppression systems showing type of installation and basis for type selected.

143 204

f. A listing showing the effect of each postulated fire identified in paragraph 1 of Section 9.5.1.1 on capability of safe reactor shutdown and potential release of radioactive material.

9.5.1.4 Inspection and Testing Requirements. The PSAR should list and discuss the installation, testing, and inspection planned during construction of the fire protection systems to demonstrate the integrity of the systems as installed. Describe in the FSAR the periodic operational checks, inspection, and servicing required to maintain this integrity. In the FSAR, discuss the periodic operational testing necessary to maintain a highly reliable alarm detection system.

9.5.1.5 Personnel Qualification and Training

1. The SAR should state the qualification requirements for the fire protection engineer or consultant who is to be responsible for the preparation of the Fire Hazards Analysis and for the design and selection of equipment; inspect and test the complete physical aspects of the system; develop the fire protection program; and assist in the firefighting training for the operating plant. In the FSAR, discuss the initial training and the updating provisions such as fire drills provided for maintaining the competence of the station firefighting and operating crew, including personnel responsible for maintaining and inspecting the fire protection equipment.

2. Administrative procedures consistent with the need for maintaining the performance of the fire protection system and personnel in nuclear power plants should be provided in the SAR.

Guidance is contained in the following National Fire Protection Association (NFPA) publications:

NFPA 4 - Organization for Fire Services

NFPA 4A - Organization of a Fire Department

NFPA 6 - Industrial Fire Loss Prevention

NFPA 7 - Management of Fire Emergencies

NFPA 8 - Management Responsibilities for Effects of Fire on Operations

NFPA 27 - Private Fire Brigades

NFPA 802 - Recommended Fire Protection Practice for Nuclear Reactors.

3. The QA programs of applicants and contractors should ensure that the guidelines for design, procurement, installation, and testing

and the administrative controls for the fire protection systems for safety-related areas are satisfied. The QA program should be under the management control of the QA organization. This control consists of (a) formulating a fire protection QA program that incorporates suitable requirements and is acceptable to the management responsible for fire protection or verifying that the program incorporates suitable requirements and is acceptable to the management responsible for fire protection and (b) verifying the effectiveness of the QA program for fire protection through review, surveillance, and audits. Performance of other QA program functions for meeting the fire protection program requirements may be performed by personnel outside the QA organization. The QA program for fire protection should be part of the overall plant QA program and should be described in the SAR.

4. The SAR should describe the applicant's Emergency Response Plan with respect to fire protection. The need for good organization, training, and equipping of fire brigades at nuclear power plants requires that effective measures be implemented to ensure proper discharge of these functions. The guidance in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," should be followed as applicable. Successful firefighting requires both proper equipment and personnel capable of using it efficiently.

a. A program of periodic maintenance and testing of fire protection systems and equipment, including emergency lighting and communication equipment, will ensure the first of these requirements.

b. Basic training is the other necessary element in effective firefighting operation. Such training can only be accomplished by conducting drills and classroom instruction several times a year so that all members of the fire brigade have had the opportunity to train as a team testing itself in the major areas of the plant.

c. To have proper coverage during all phases of operation, members of each shift crew should be trained in fire protection. Training of the plant fire brigade should be coordinated with the local fire department so that responsibilities and duties are delineated in advance. This coordination should be part of the training course and should be included in the training of the local fire department staff. The plant fire brigade should not include any of the plant physical security personnel required to be available to fulfill the response requirements of paragraph 73.55(h)(2) of 10 CFR Part 73, "Physical Protection of Plants and Materials." Local fire departments should be provided training in operational precautions when fighting fires on nuclear power plant sites and should be made aware of the need for radiological protection of personnel and the special hazards associated with a nuclear power plant site.

9.5.2 Communication Systems

9.5.2.1 Design Bases. The design bases for the communication systems for intra-plant and plant-to-offsite communications should be provided and should include a discussion of the use of diverse system types.

143 206

9.5.2.2 System Description. A description and evaluation of the communication systems should be provided. The FSAR should provide a detailed description and drawings.

9.5.2.3 Inspection and Testing Requirements. The inspection and testing requirements for the communication systems should be provided.

9.5.3 Lighting Systems

A description of the normal lighting system for the plant should be provided. A description of the emergency lighting system, including design criteria and a failure analysis, should also be provided.

9.5.4 Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 Design Bases. The design bases for the fuel oil storage and transfer system for the diesel generator should be provided and should include the requirement for onsite storage capacity, capability to meet design criteria (e.g., single-failure criterion), code design requirements, and environmental design bases.

A description of the diesel generator fuel oil storage and transfer system, including drawings, should be provided in the PSAR. The FSAR should provide a detailed description and drawings.

An evaluation of the fuel oil storage and transfer system should be provided and should include the potential for material corrosion and fuel oil contamination, a failure analysis to demonstrate capability to meet design criteria (e.g., single-failure criterion), ability to withstand environmental design conditions, and the plans by which additional oil may be procured, if required.

9.5.5 Diesel Generator Cooling Water System

The design bases for the cooling water system should be provided and should include a discussion of the ability to meet the single-failure criterion. A description of the cooling water system, including drawings, should be provided.

9.5.6 Diesel Generator Starting System

The design bases for the starting system, including required system capacity, should be provided and should include a discussion of the ability to meet the single-failure criterion. A description of the starting system, including drawings, should be provided.

9.5.7 Diesel Generator Lubrication System

The design bases for the lubrication system should be provided and should include a discussion of the ability to meet the single-failure

143 207

criterion. A description of the lubrication system, including drawings, should be provided.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

9.5.8.1 Design Bases. This section should provide the design bases for the diesel generator combustion air intake and exhaust system, including the bases for protection from the effects of natural phenomena, missiles, and contaminating substances as related to the facility site, systems, and equipment and the capability of the system to meet minimum safety requirements assuming a single failure. Seismic and quality group classifications should be provided in Section 3.2 and referenced in this section.

9.5.8.2 System Description. A complete description of the system should be provided, including system drawings detailing component redundancy, where required, and showing the location of system equipment in the facility and the relationship to site systems or components that could affect the system.

9.5.8.3 Safety Evaluation. Analyses should be provided to demonstrate that the minimum quantity and oxygen content requirements for intake combustion air will be met considering such effects as recirculation of diesel combustion products, accidental release of gases stored in the vicinity of the diesel intakes, restriction of inlet airflow, intake of such particulates as airborne dust, and low barometric pressure. The results of failure mode and effects analyses to ensure minimum requirements should be provided. If system degradation could result from the consequences of missiles or failures of high- or moderate-energy piping systems located in the vicinity of the combustion air intake and exhaust system, assurance should be provided that such degradation would not jeopardize the system's minimum safety functional requirements.

9.5.8.4 Inspection and Testing Requirements. Inspection and periodic system testing requirements for the diesel generator combustion air intake and exhaust system should be described.

143 208

10. STEAM AND POWER CONVERSION SYSTEM

This chapter of the SAR should provide information concerning the plant steam and power conversion system. For purposes of this chapter, the steam and power conversion system (heat utilization system) should be considered to include the following:

1. The steam system and turbine generator units of an indirect-cycle reactor plant, as defined by the secondary coolant system, or

2. The steam system and turbine generator units in a direct-cycle plant, as defined by the system extending beyond the reactor coolant system isolation valves.

There will undoubtedly be many aspects of the steam portion of the plant that have little or no relationship to protection of the public against exposure to radiation. The SAR is, therefore, not expected to deal with this part of the plant to the same depth or detail as those features playing a more significant safety role. Enough information should be provided to allow understanding in broad terms of what the secondary plant (steam and power conversion system) is, but emphasis should be on those aspects of design and operation that do or might affect the reactor and its safety features or contribute toward the control of radioactivity. The capability of the system to function without compromising directly or indirectly the safety of the plant under both normal operating or transient situations should be shown by the information provided. Where appropriate, the evaluation of radiological aspects of normal operation of the steam and power conversion system and subsystems should be summarized in this chapter and presented in detail in Chapters 11 and 12.

10.1 Summary Description

A summary description indicating principal design features of the steam and power conversion system should be provided. An overall system flow diagram and a summary table of the important design and performance characteristics, including a heat balance at rated power and at stretch power, should be provided. The description should indicate those system design features that are safety related.

10.2 Turbine-Generator

10.2.1 Design Bases

The design bases for the turbine-generator equipment should be provided and should include the performance requirements under normal, upset, emergency, and faulted conditions; intended mode of operation (base loaded or load following); functional limitations imposed by the design or operational characteristics of the reactor coolant system (rate at which electrical load may be increased or decreased with and without reactor control rod motion or steam bypass); and design codes to be applied.

10.2.2 Description

A description of the turbine-generator equipment, including moisture separation, use of extraction steam for feedwater heating, and control functions that could influence operation of the reactor coolant system, should be provided as well as drawings. The turbine-generator-overspeed control system should be described in detail, including redundancy of controls, type of control utilized, overspeed setpoints, and valve actions required for each setpoint.

10.2.3 Turbine Disk Integrity

The failure of a turbine disk or rotor might produce a high-energy missile that could damage a safety-related component. This section should provide information to demonstrate the integrity of turbine disks and rotors.

10.2.3.1 Materials Selection. This section should include materials specifications, fabrication history, and chemical analysis of the disk and rotor forgings. Particular attention should be paid to items affecting fracture toughness and metallurgical stability. The mechanical properties of the disk material such as yield strength and fracture toughness should be listed. The methods of obtaining these properties should be described.

10.2.3.2 Fracture Toughness. The criteria used to ensure protection against brittle failure of low-pressure turbine disks should be described. Include detailed information on ductile-brittle transition temperature (NDT or FATT) and minimum operating temperature. If a fracture mechanics approach is used, the analytical method and the key assumptions made should be described.

10.2.3.3 High-Temperature Properties. Provide the stress-rupture properties of the high-pressure rotor material, and describe the method for obtaining these properties.

10.2.3.4 Turbine Disk Design. Provide the following design information for low-pressure disks and high-pressure rotors:

1. The tangential stress due to centrifugal loads, interference fit, and thermal gradients at the bore region at normal speed and design overspeed.

2. The maximum tangential and radial stresses and their location.

10.2.3.5 Preservice Inspection. Describe the preservice inspection procedures and acceptance criteria to demonstrate the initial integrity of the disks and rotors.

10.2.3.6 Inservice Inspection. The inservice inspection program for the turbine assembly and the inspections and tests of the main steam stop and control valves and the reheat stop and intercept valves should be described.

10.2.4 Evaluation

An evaluation of the turbine-generator and related steam handling equipment should be provided. This evaluation should include a summary discussion of the anticipated operating concentrations of radioactive contaminants in the system, radiation levels associated with the turbine components and resulting shielding requirements, and the extent of access control necessary based on radiation levels and shielding provided. Details of the radiological evaluation should be provided in Chapters 11 and 12.

10.3 Main Steam Supply System

10.3.1 Design Bases

The design bases for the main steam line piping from the steam generator, in the case of an indirect cycle plant, or from the outboard isolation valve, in the case of a direct cycle plant, should be provided and should include performance requirements, environmental design bases, inservice inspection requirements, and design codes to be applied. Capability of the system to dump steam to the atmosphere, if required, should be discussed. Steam lines to and from feedwater turbines should be included in the descriptions.

10.3.2 Description

A description of the main steam line piping, including drawings showing interconnected piping, should be provided.

10.3.3 Evaluation

An evaluation of the design of the main steam line piping should be provided and should include an analysis of the ability to withstand limiting environmental and accident conditions and provisions for permitting inservice inspections to be performed. Appropriate references should be made to seismic classifications in Chapter 3 and to the analysis of postulated high-energy line failure in Section 3.6.

10.3.4 Inspection and Testing Requirements

The inspection and testing requirements of the main steam line piping should be described. Describe the proposed requirements for preoperational and inservice inspection of steam line isolation valves or reference other sections of the SAR where these are described.

143 211

10.3.5 Water Chemistry (PWR)

The effect of the water chemistry chosen on the radioactive iodine partition coefficients in the steam generator and air ejector should be discussed.

Detailed information on the secondary-side water chemistry, including methods of treatment for corrosion control and proposed specification limits should be provided. Discuss methods for monitoring and controlling water chemistry.

10.3.6 Steam and Feedwater System Materials

This section should provide the information indicated below on the materials used for Class 1, 2, and 3 components.

10.3.6.1 Fracture Toughness. Indicate the degree of compliance with the test methods and acceptance criteria of the ASME Code Section III in Articles NB-2300, NC-2300, and ND-2300 for fracture toughness for ferritic materials used in Class 1, 2, and 3 components.

10.3.6.2 Materials Selection and Fabrication. Information on materials selection and fabrication methods used for Class 1, 2, and 3 components should include the following:

1. For any material not included in Appendix I to Section III of the ASME Code or in Regulatory Guide 1.85, "Materials Code Case Acceptability--ASME Section III Division 1," provide the data called for under Appendix IV for approval of new materials. The use of such materials should be justified.

2. For austenitic stainless steel components, the degree to which the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel;" Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel;" and Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," are followed should be indicated. Justification for any deviations from the procedures shown in these guides should be provided.

3. Information on the cleaning and handling of all Class 1, 2, and 3 components should be provided. The degree to which the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and ANSI N45.2.1-73, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Plants," are followed should be indicated. Justification for any deviations from the position in these documents should be provided.

4. Indicate whether the preheat temperatures used for welding low-alloy steel are in accordance with Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." Justification for any deviations from the procedures shown in this guide should be provided.

143 212

For carbon steel materials, indicate whether the preheat temperatures are in accordance with Section III, Article D-1000, of the ASME Boiler and Pressure Vessel Code. The use of alternative procedures should be justified.

5. For all applicable components, the degree to which the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," are followed should be indicated. Justification for any deviations from the procedures given in this guide should be provided.

6. Indicate that the nondestructive examination procedures used for the examination of tubular products conform to the requirements of the ASME Boiler and Pressure Vessel Code. Provide justification for deviations from these requirements.

10.4 Other Features of Steam and Power Conversion System

This section of the SAR should provide discussions of each of the principal design features and subsystems of the steam and power conversion system. Because these systems vary in number, type and nomenclature for various plant designs, the Standard Format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 10.4.1 through 10.4.X) for each. These subsections should provide the following information:

1. Design bases,

2. System description,

3. Safety evaluation,

4. Tests and inspections, and

5. Instrumentation applications for each subsystem or feature.

The following paragraphs provide examples of subsystems and features that should be discussed, as appropriate to the individual plant, and identify some specific information that should be provided in addition to the items identified above.

10.4.1 Main Condensers

The description of the main condensers should include performance requirements, materials of construction, methods used to reduce the probability of corrosion and/or erosion of tubes and components, anticipated inventory of radioactive contaminants during power operation and during shutdown, anticipated air leakage limits, control functions that could influence operation of the primary reactor coolant or secondary systems, potential for hydrogen buildup, and provisions for protection of safety-related equipment from flooding as a result of failure of the condenser.

143 213

The description should also include the methods used to detect the leakage of cooling water into the condensate, the contaminants allowed in the condensate, the procedure to repair condensate leaks, and the length of time the condenser may operate with degraded conditions without affecting the condensate/feedwater quality for safe operation.

10.4.2 Main Condenser Evacuation System

The description of the evacuation systems for the main condensers should include performance requirements for startup and normal operation, anticipated release rates of radioactive materials, evaluation of the capability to limit or control loss of radioactivity to the environment, and control functions that could influence operation of the reactor coolant system. Describe any design features that preclude the existence of explosive mixtures. Details of the radiological evaluation should be provided in Chapter 11.

10.4.3 Turbine Gland Sealing System

The discussion of the turbine gland sealing system should include identification of the source of noncontaminated steam, a description of potential radioactivity leakage to the environment in the event of a malfunction, and discussion of the means to be used to monitor system performance. The inspection and testing requirements should be described. The evaluation of the estimate of potential radioactivity leakage to the environment in the event of a malfunction of the turbine gland sealing system should be provided in Chapter 15. Details of the radiological evaluation should be provided in Chapter 11.

10.4.4 Turbine Bypass System

The design bases for the turbine bypass system should include performance requirements, capability to meet design criteria, design codes to be applied, and environmental criteria. The evaluation of the turbine bypass system should include a failure analysis to determine the effect of equipment malfunctions on the reactor coolant system.

10.4.5 Circulating Water System

The description of the circulating water system should include discussion of performance requirements; dependence on the system for cooling during shutdown; anticipated operational occurrences and accidents; control of the circulating water chemistry, corrosion, and organic fouling; environmental influences; and potential interaction of cooling towers, if any, with the plant structure. The potential for flooding safety-related equipment due to the failure of a system component such as an expansion joint should be discussed. References to paragraphs 2.4.11.5 and 2.4.11.6 should be provided, where applicable.

10.4.6 Condensate Cleanup System

The design bases for the condensate cleanup system should include the fraction of condensate flow to be treated, impurity levels to be maintained, and design codes to be applied. The evaluation of the condensate cleanup system should include an analysis of demineralizer capacity and anticipated impurity levels, an analysis of the contribution of impurity levels from the secondary system to reactor coolant system activity levels, and performance monitoring. Provisions for the control of chloride ion and other contaminants should be described.

10.4.7 Condensate and Feedwater Systems

The design bases for the condensate and feedwater systems should include design codes to be applied, criteria for isolation from the steam generator or reactor coolant system, supply of condensate available for emergency purposes, inservice inspection requirements, and environmental design requirements. The evaluation of the condensate and feedwater systems should include an analysis of component failure, effects of equipment malfunction on the reactor coolant system, and an analysis of isolation provisions to preclude release of radioactivity to the environment in the event of a pipe leak or break.

Provide the following information with reference to fluid flow instabilities, e.g., water hammer, for steam generators using top feed:

1. A description of normal operating transients that could cause the water level in the steam generator to drop below the sparger or cause the nozzles to uncover and allow steam to enter the sparger and feedwater piping.

2. A summary of the criteria for routing or isometric drawings showing the routing of the feedwater piping system from the steam generators to the restraint that is closest, on the upstream side, to the feedwater isolation valve that is outside containment.

3. A description of the piping system analyses, including any forcing functions, or the result of test programs performed to verify that uncovering of feedwater lines could not occur or that such uncovering would not result in unacceptable damage to the system. A summary of relevant water hammer experience is provided in NUREG-0291, "An Evaluation of PWR Steam Generator Water Hammer."

10.4.8 Steam Generator Blowdown System (PWR)

10.4.8.1 Design Bases. This section should provide the design bases for the steam generator blowdown system (SGBS) in terms of its ability to maintain optimum secondary-side water chemistry in recirculating steam generators of PWRs during normal operation, including anticipated operational occurrences (main condenser inleakage and primary-to-secondary leakage). The design bases should include consideration of expected and

143 215

design flows for all modes of operation (process and process bypass), process design parameters and equipment design capacities, expected and design temperatures for temperature-sensitive treatment processes (demineralization and reverse osmosis), and process instrumentation and controls for maintaining operations within established parameter ranges.

Seismic and quality group classifications of the SGBS should be provided in Section 3.2 and referenced in this section.

10.4.8.2 System Description and Operation. A detailed description of the SGBS, including component description, piping and instrumentation diagrams, process flow diagrams, and equipment general arrangement drawings (reference may be made to pertinent information in Section 11.2), should be provided. Discuss the operating procedures and the processing to be provided for all anticipated modes of operation, including system or process bypass, significant primary-to-secondary leakage, and main condenser inleakage.

Discuss the instrumentation and controls provided to protect temperature-sensitive elements (demineralizer resins or reverse osmosis membranes) and to control flashing, liquid levels, and process flow through system components. The radioactive waste treatment and process and effluent radiological monitoring aspects of the SGBS should be described in Sections 11.2, 11.3, 11.4, and 11.5.

10.4.8.3 Safety Evaluation. The interfaces between the SGBS and other plant systems should be discussed. Unusual design conditions that could lead to safety problems should be identified and evaluated. Provide a failure mode and effects analysis of any interactions that may incapacitate safety-related equipment. Provide coolant chemistry specifications to demonstrate compatibility with primary-to-secondary system pressure boundary material. The bases for the selected chemistry limits should be included. (Information provided in Section 5.4.2 may be referenced.)

10.4.8.4 Tests and Inspections. The inspection and periodic testing requirements for the SGBS should be described.

10.4.9 Auxiliary Feedwater System (PWR)

10.4.9.1 Design Bases. This section should provide the design bases for the auxiliary feedwater system in terms of the safety-related functional performance requirements of the system, including the required pumping capacities of the pumps, diversity of power supplied to the system pumps and system control valves, capabilities of the pumps (head/flow) with respect to supply requirements of the steam generator, and the auxiliary feedwater supply capacity requirements for makeup during maximum hot standby conditions and for cold shutdown of the facility following a reactor trip or accident condition; requirement for the system's ability to withstand adverse environmental occurrences and the effects of pipe breaks; requirement of the system to perform its safety-related function in the

143 216

event of a single failure coincident with pipe breaks, environmental occurrences, and the loss of offsite power and/or the standby a.c. power system. The means by which the system is protected from the effects of hydraulic instability (water hammer) or the design considerations precluding the occurrence of hydraulic instability should be provided.

Seismic and quality group classification should be provided in Section 3.2 and referenced in this section.

10.4.9.2 System Description. A detailed description of the auxiliary feedwater system should be provided, including piping and instrumentation diagrams, system drawings, and the location of components in the station complex. The description and drawings should also include subsystems, system interactions, components utilized, piping connection points, instrumentation and controls utilized, and system operations, i.e., system function during normal operations and the minimum functional conditions of the system in the event of pipe breaks, loss of main feedwater system, or loss of offsite power. The information should also state the maximum length of time the plant could do without normal feedwater and the minimum auxiliary feedwater flow rate required after this time period (i.e., pumps started and control valves open) for these conditions.

10.4.9.3 Safety Evaluation. An evaluation of the capability of the auxiliary feedwater system should include (either in this section or by reference) the means by which protection from postulated failures of high- and moderate-energy systems is accomplished for the system and auxiliary supporting systems and the means by which the system is capable of withstanding the effects of site-related natural phenomena. Failure mode and effects analyses should be provided that ensure minimum safety requirements are met assuming a postulated pipe failure concurrent with a single active component failure in any system required to ensure performance of the auxiliary feedwater system. An analysis should demonstrate the capability of the system to preclude hydraulic instabilities (characterized as water hammer) from occurring for all modes of operation.

An analysis or analyses to demonstrate the system's capability to perform its safety function when subjected to a combination of environmental occurrences, environmental conditions, pipe break, and loss of power during normal and accident conditions should be performed. In addition, an analysis should be performed to demonstrate the system's capability to perform its safety function utilizing diverse power sources to ensure system operability without reliance on a.c. power.

10.4.9.4 Inspection and Testing Requirements. The inspection and periodic testing requirements for the auxiliary feedwater system should be described.

10.4.9.5 Instrumentation Requirements. The system instrumentation and controls should be described. The adequacy of safety-related instrumentation and controls to fulfill their functions should be demonstrated.

143 217

11. RADIOACTIVE WASTE MANAGEMENT

This chapter should describe:

1. The capabilities of the plant to control, collect, handle, process, store, and dispose of liquid, gaseous, and solid wastes that may contain radioactive materials, and

2. The instrumentation used to monitor the release of radioactive wastes.

The information should cover normal operation, including anticipated operational occurrences (refueling, purging, equipment downtime, maintenance, etc.). The proposed radioactive waste (radwaste) treatment systems should have the capability to meet the requirements of 10 CFR Parts 20 and 50 and the recommendations of appropriate regulatory guides concerning system design, control and monitoring of releases, and maintaining releases of radioactive materials at the "as low as is reasonably achievable" level in accordance with Appendix I to 10 CFR Part 50.

11.1 Source Terms

The PSAR should indicate the sources of radioactivity that serve as design bases for the various radioactive waste treatment systems for normal operation, including anticipated operational occurrences, as well as for design conditions. The parameters used to determine the specific activity of each radioisotope in the primary and secondary (PWR) coolant should be described and all assumptions justified.

The PSAR should provide the concentrations of fission, activation, and corrosion products used in the source term calculations and their bases. The activation of water and constituents normally found in the reactor coolant system should also be taken into account. The source of each isotope (e.g., C-14, Ar-41) should be identified and the concentration of each isotope indicated. Provide the basis for the values used. Previous pertinent operating experience should be cited.

Mathematical models and parameters used to calculate source terms for normal operation, including anticipated operational occurrences, should be provided.

For the purpose of evaluating the adequacy of various ventilation systems, provide in the PSAR estimates of the leakage rate from the reactor coolant system and other fluid systems containing radioactivity into individual cubicles and areas that may require access by operating personnel. Tabulate the sources of leakage. Estimates of the releases of radioactive gases and radioiodines from each leakage source and their subsequent transport and release path should be provided. The basis for the values used should be indicated. Cite previous pertinent experience from operating reactors. Discuss leakage measurements and special design features to

143 218

reduce leakage. The principal discussions of coolant leakage in other sections of the SAR should be referenced.

The PSAR should identify all sources of releases of radioactive material that are not normally considered part of the radioactive waste management systems, e.g., the steam generator blowdown system, building ventilation exhaust systems, containment purging, and the turbine gland seal system. Estimates of the release of radioactive materials (by radionuclide) from each source identified and the subsequent transport mechanism and release path should be provided. Identify planned operations, including anticipated operational occurrences, that may result in release of radioactive materials to the environment. Consider leakage rates and concentrations of radioactive materials for both expected and design conditions. The bases for all values used should be provided. Describe changes from previous designs that may affect the release of radioactive materials to the environment.

The FSAR should provide additional information required to update the PSAR to the final design conditions.

11.2 Liquid Waste Management Systems

This section should describe the capabilities of the plant to control, collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences. Process and effluent radiological monitoring and sampling systems should be described in Section 11.5.

11.2.1 Design Bases

The PSAR should provide the design objectives and design criteria for the liquid radioactive waste handling and treatment systems in terms of expected annual quantities of radioactive material (by radionuclide) released, averaged over the life of the plant, and the expected doses to individuals at or beyond the site boundary. An evaluation should be included to show that the proposed systems are capable of controlling releases of radioactive materials within the numerical design objectives of Appendix I to 10 CFR Part 50. The evaluation should also show that the proposed systems contain all items of reasonably demonstrated technology that, when added to the system and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. All assumptions should be provided and the calculational methods should be shown. An evaluation should be provided to show that the proposed systems have sufficient capacity, redundancy, and flexibility to meet the concentration limits of 10 CFR Part 20 during periods of equipment downtime and during operation at design basis fuel leakage (i.e., leakage from fuel producing 1 percent of the reactor power for a PWR or fuel having a noble gas release rate of 100 μCi/sec per MWt after a 30-minute decay for a BWR).

143 219

A tabulation showing the liquid radwaste system components and their design parameters, e.g., flow, temperature, pressure, and materials of construction, should be provided. An evaluation indicating the capabilities of the system to process surge waste flows associated with anticipated operational occurrences such as anticipated waste flows from back-to-back refueling and equipment downtime should be included.

The seismic design criteria and analytical procedures for structures housing the liquid radwaste components should be provided along with the quality group classification for the liquid radwaste components and piping. Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference. The PSAR should describe how the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50 will be implemented.

Design features incorporated to reduce maintenance, equipment downtime, liquid leakage, or gaseous releases of radioactive materials to the building atmosphere or to facilitate cleaning or otherwise improve radwaste operations should be described.

The design provisions incorporated to control the release of radioactive materials due to overflows from all liquid tanks outside containment that could potentially contain radioactive materials should be described. Discuss the effectiveness of both the physical and the monitoring precautions taken, e.g., dikes, level gauges, and automatic diversion of wastes from tanks exceeding a predetermined level. The potential for operator error or equipment malfunctions (single failures) to result in uncontrolled releases to the environment should be discussed. Describe the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity to the environs. Process and effluent radiological monitoring systems should be described in Section 11.5.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.2.2 System Description

The PSAR should include a description of each liquid waste subsystem and the process flow diagrams indicating processing equipment, normal process routes, equipment capacities, and redundancy in equipment. For multi-unit stations, those subsystems that are shared should be indicated. All equipment and components that will normally be shared between subsystems should be identified. Indicate the processing to be provided for all liquid radwastes, including turbine building floor drains and steam generator blowdown liquids (PWR).

For each subsystem, tabulate or show on the flow diagrams the maximum and expected inputs in terms of flow (gal/day per reactor) and radioactivity (fraction of primary coolant activity) for normal operation, including anticipated operational occurrences. The bases for the values used should be provided.

The segregation of liquid waste streams based on conductivity, radioactivity, and chemical composition, as appropriate, should be described. Indicate all potential bypass routes, the conditions governing their use, and their anticipated frequency of bypass due to equipment downtime. The piping and instrumentation diagrams (P&IDs) should indicate system interconnections and seismic and quality group interfaces.

The location of secondary flow paths for each system should be indicated. The normal operation of each system and differences in system operation during anticipated operational occurrences such as startups, shutdowns, and refueling should be described.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.2.3 Radioactive Releases

The PSAR should provide the criteria for determining whether processed liquid wastes will be recycled for reuse or further treatment or discharged to the environment. Discuss the influence the plant water balance (requirements) and the expected tritium concentrations in process streams will have on the release parameters assumed.

The parameters and assumptions used to calculate releases or radioactive materials in liquid effluents and their bases should be provided. Provide the expected releases of radioactive materials (by radionuclides) in liquid effluents resulting from normal operation, including anticipated operational occurrences, and from design bases fuel leakage in Ci/yr per reactor.

Tabulate the releases by radionuclide for each subsystem and for the total system, and indicate the effluent concentrations. The calculated effluents should be compared with the concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 2; the doses due to the effluents should be compared with the numerical design objectives of Appendix I to 10 CFR Part 50 and the dose limits of 10 CFR Part 20. Identify all release points for liquid wastes and the dilution factors considered in the evaluation.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.3 Gaseous Waste Management Systems

This section should describe the capabilities of the plant to control, collect, process, handle, store, and dispose of gaseous radioactive waste generated as the result of normal operation and anticipated operational occurrences. Process and effluent radiological monitoring systems should be described in Section 11.5.

143 221

In this section, the term "gaseous waste systems" is applied to all plant systems that have a potential to release radioactive materials in gaseous effluent to the environment, including building ventilation systems. Gaseous wastes include noble gases, halogens, tritium, argon-41, carbon-14, and radioactive material in particulate form.

11.3.1 Design Bases

The PSAR should provide the design objectives and design criteria for the gaseous radioactive waste handling and treatment systems in terms of expected annual quantities of radioactive material (by radionuclide) released, averaged over the life of the plant, and expected doses to individuals at or beyond the site boundary. An evaluation should be provided to show that the proposed systems are capable of controlling releases of radioactive materials within the numerical design objectives of Appendix I to 10 CFR Part 50. The evaluation should also show that the proposed systems contain all items of reasonably demonstrated technology that, when added to the system and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. All assumptions should be provided and the calculational methods should be shown. An evaluation should be provided to show that the proposed systems have sufficient capacity, redundancy, and flexibility to meet the concentration limits of 10 CFR Part 20 when operating at design basis fuel leakage (i.e., leakage from fuel producing 1 percent of the reactor power for a PWR or fuel having a noble gas release rate of 100 μCi/sec per MWt after a 30-minute decay for a BWR).

The gaseous radwaste system components and their design parameters, e.g., flow, temperature, pressure, and materials of construction, should be listed. Provide an evaluation indicating the capabilities of the system to process surges in waste flows associated with anticipated operational occurrences such as cold startups, shutdowns, purging of containment, back-to-back refueling, and equipment downtime.

The seismic design criteria and analytical procedures for equipment support elements and structures housing the gaseous waste treatment system should be provided along with the quality group classification for the gaseous waste treatment components and piping. The PSAR should describe how the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50 will be implemented.

Design features incorporated to reduce maintenance, equipment downtime, leakage, and gaseous releases of radioactive materials to the building atmosphere or to facilitate cleaning or otherwise improve radwaste operations should be described.

The design provisions incorporated to control the release of radioactive materials in gaseous effluents as the result of equipment malfunction or operator error should be described. Discuss the effectiveness of monitoring precautions taken, i.e., automatic termination of waste release from waste gas storage tanks when the release exceeds a predetermined level. The potential for an operator error or equipment malfunction (single failures) that may result in uncontrolled releases of radioactivity to the environment should be discussed. Process and effluent radiological monitoring systems should be described in Section 11.5.

The design objectives of the plant ventilation systems for normal and emergency operation, including anticipated operational occurrences, should be described with respect to meeting the requirements of 10 CFR Parts 20 and 50.

For systems where the potential for an explosion exists, any equipment that is not designed to withstand the pressure peak of the explosion should be identified and justification provided. Process instrumentation (including gas analyzers) and design features provided to prevent explosions should be described along with provisions to ensure that seals will not be permanently lost following an explosion.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.3.2 System Description

The PSAR should include a description of each gaseous waste subsystem and the process flow diagrams indicating processing equipment, normal flow paths through the system, equipment capacities, and redundancy in equipment. For multi-unit stations, those subsystems that are shared should be indicated. All equipment and components that will normally be shared between subsystems should be identified. For each subsystem, tabulate or show on the flow diagrams the maximum and expected inputs in terms of flow and radioactivity content for normal operation, including anticipated operational occurrences. The bases for the values used should be provided. Indicate the composition of carrier and blanket gases, and describe the segregation of streams containing hydrogen, if appropriate.

The piping and instrumentation diagrams should indicate system interconnections and seismic and quality group interfaces. Instrumentation and controls that govern the operation should be described. Indicate all potential bypasses of normal process routes, the conditions governing their use, and the anticipated frequency of bypass due to equipment downtime. Provide the location of liquid seals, show them on the P&IDs, and describe how blown seals will be automatically reestablished. The location of vents and secondary flow paths for each system should be indicated. Describe both the normal operation of each system and the differences in system operation during anticipated operational occurrences such as startups, shutdowns, refueling, and purging of containment.

The ventilation system for each building that can be expected to contain radioactive materials should be described. Include building volumes, expected flow rates from buildings and equipment cubicles, filter characteristics, and the design criteria on which these are based. Describe

145 223

both the normal operation of each ventilation system and the differences in operation during anticipated operational occurrences such as startup, shutdown, and refueling. Chapter 9 should be referenced, as appropriate. The FSAR should provide a tabulation showing the calculated concentrations of airborne radioactive material (by radionuclide) expected during normal and anticipated operational occurrences for equipment cubicles, corridors, and areas normally occupied by operating personnel.

The subsystems in the steam and power conversion systems that are potential sources of gaseous radioactive effluents should be described. Examples of such systems are the turbine gland sealing systems and the main condenser vacuum system. Provide the flow rates and concentrations of radioactive materials (by radionuclide) through these systems during normal operations and anticipated operational occurrences. The bases for the values used should be provided. Tabulate the expected frequency and quantity of steam released during steam dumps to the atmosphere (PWR) or pressure relief valve venting to the suppression pool (BWR). The bases for the values used should be provided. Other sections of the SAR should be referenced, as appropriate.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.3.3 Radioactive Releases

The PSAR should provide the criteria to be used for releasing gaseous wastes and the acceptable release rates.

The parameters and assumptions used in calculating releases of radioactive materials in gaseous effluents and their bases should be provided. Provide the expected releases of radioactive materials (by radionuclides) in gaseous effluents resulting from normal operation, including anticipated operational occurrences, in Ci/yr per reactor.

Tabulate the releases by radionuc[illegible] each subsystem and for the total system, and indicate the efflu[illegible] [illegible]entrations. The calculated effluents should be compared with the co[illegible]entration limits of 10 CFR Part 20, Appendix B, Table II, Column 1; the doses due to the effluents should be compared with the numerical design objectives of Appendix I to 10 CFR Part 50 and the dose limits of 10 CFR Part 20. The dilution factors considered in the evaluation should be indicated.

Identify all release points of gaseous waste to the environment on process flow diagrams, general arrangement drawings, or a site plot plan.

For [illegible] points, give:

1. Height of release,

2. Inside dimensions of release point exit,

143 224

3. Effluent temperature, and

4. Effluent exit velocity.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.4 Solid Waste Management System

This section should describe the capabilities of the plant to control, collect, handle, process, package, and temporarily store prior to shipment wet and dry solid radioactive waste generated as a result of normal operation, including anticipated operational occurrences. In this section, the term "solid waste management system" means a permanently installed system. Process and effluent radiological monitoring systems should be described in Section 11.5.

11.4.1. Design Bases

The PSAR should provide the design objectives and design criteria for the solid radioactive waste handling and treatment system in terms of the types of wastes, the maximum and expected volumes to be handled, and the isotopic and curie content. The seismic design criteria and analytical procedures for structures housing the solid radwaste system should be provided along with the quality group classification for the solid radwaste components and piping. Seismic and quality group classifications provided in Section 3.2 may be incorporated by reference. Indicate how the requirements of 10 CFR Parts 20, 50, and 71, and applicable DOT regulations will be implemented.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.4.2 System Description

The PSAR should describe the wet solid waste subsystem to be used for processing ion exchange resins, filter sludges, evaporator bottoms, and miscellaneous liquids. List the system components (evaporator concentrates, sludges tanks, phase separator tanks, etc.). Their design capacity and materials of construction should be indicated. In the PSAR, tabulate the maximum and expected waste inputs, their physical form (resin, sludge, etc.), sources of waste, volume per batch, and isotopic composition. The bases for the values used should be provided. Describe the method to be used for solidifying each waste type, the type of container in which the wastes will be packaged, and the means to be used to ensure the absence of free liquid in the waste containers, including in the FSAR the process control program to ensure a solid matrix.

Process flow diagrams indicating the normal process route, flow rates, equipment holdup times, expected isotopic content of each flow, and equipment capacities should be provided. Describe the instrumentation and controls

143 225

used for process control. Provide the piping and instrumentation diagrams that show system interconnections and seismic and quality group interfaces. Describe the design provisions incorporated to control the release of radioactive materials due to overflows from tanks containing liquids, sludges, and spent resins. Identify all tanks or equipment that use compressed gases for any function and provide information as to the gas flow rate volume per operation, expected number of operations per year, expected radionuclide concentration of offgases, treatment provided, and interfaces with ventilation exhaust systems. Discuss the effectiveness of the physical and monitoring precautions taken (e.g., retention basins, curbing, and level gauges). The potential for operator errors or equipment malfunctions (single failures) that may result in uncontrolled releases of radioactive material should be discussed.

Describe the dry solid waste subsystem to be used for processing dry filter media (ventilation filters); contaminated clothing, equipment, tools, and glassware, and miscellaneous radioactive wastes that are not amenable to solidification prior to packaging. Tabulate the maximum and expected waste inputs in terms of type (filters, tools, etc.), sources of waste, volume, and isotopic and curie content. The bases for the values used should be provided. Describe the method of packaging and equipment to be used. The provisions to be used to control airborne radioactivity due to dust during compaction and baling operations should be described. Discuss the methods of handling and packaging large waste materials and equipment that has been activated during reactor operation (e.g., core components).

Describe the containers to be used for packaging wastes and indicate their compliance with applicable Federal regulations. Provisions for sealing, decontaminating, and moving the containers to storage and to shipping areas should be discussed along with the potential for radioactive spills due to dropping containers from cranes, monorails, etc. Describe provisions for collecting and processing decontamination liquids and spillage. The provisions for waste storage prior to shipping, including the storage capacity and the expected onsite storage time should be described. Layout drawings of the packaging, storage, and shipping areas should be provided.

The maximum and expected annual volumes and the curie and isotopic content of wastes to be shipped offsite for each waste category should be indicated.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

This section should describe the systems that monitor and sample the process and effluent streams in order to control releases of radioactive materials generated as the result of normal operations, including anticipated operational occurrences, and during postulated accidents. The process sampling system should be described in Section 9.3.2.

11.5.1 Design Bases

The PSAR should include the design objectives and design criteria for the process and effluent radiological monitoring systems and the sampling systems in relation to the requirements of 10 CFR Parts 20 and 50. Indicate whether, and if so how, the guidance of Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," will be followed; if it will not be followed, the specific alternative approaches to be used should be described. For the effluent monitoring system, distinguish between the design objectives for normal operations, including anticipated operational occurrences, and the design objectives for monitoring postulated accidents.

The FSAR should provide any additional information required to update the PSAR to the final design conditions.

11.5.2 System Description

Provide system descriptions for radiation detectors and samplers used to monitor and control releases of radioactive materials generated as the result of normal operations, including anticipated operational occurrences, and during postulated accidents.

For continuous process and effluent radiation monitors, provide the following information:

1. Location of monitors,

2. Type of monitor, sensitivity,* and measurement made (e.g., gross, β-γ, or isotopic analysis),

3. Instrumentation, redundancy, independence, and diversity of the components supplied,

4. Range of radioactivity concentrations to be monitored and bases for range provided,

5. Types and locations of annunciators, alarms, and automatic controls and actions initiated by each,*

6. Provisions for emergency power supplies,

7. Setpoints for alarms and controls and bases for values chosen,* and

*FSAR only.

143 227

8. Description of provisions for radiological monitoring instrument calibration, maintenance, inspection, decontamination, and replacement.*

For each location subject to routine sampling, indicate whether, and if so how, the guidance of Regulatory Guide 1.21 will be followed; if it will not be followed, the specific alternative approaches to be used should be described. The following information should be provided:

1. Basis for selecting the location,

2. Expected flow, composition, and concentrations,

3. Quantity to be measured (e.g., gross, β-γ, or isotopic concentrations),

4. Sampling frequency, type of sample nozzle or other sample equipment, and procedures used to obtain representative samples,* and

5. Analytical procedure and sensitivity.*

11.5.3 Effluent Monitoring and Sampling

Indicate how the requirements of General Design Criterion 64 will be implemented with respect to effluent discharge paths for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

11.5.4 Process Monitoring and Sampling

Indicate how the requirements of General Design Criterion 60 will be implemented with respect to the automatic closure of isolation valves in gaseous and liquid effluent discharge paths. Indicate how the requirements of General Design Criterion 63 will be implemented with respect to the monitoring of radiation levels in radioactive waste process systems.

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*FSAR only.

143 228

12. RADIATION PROTECTION

This chapter of the SAR should provide information on methods for radiation protection and on estimated occupational radiation exposures to operating and construction personnel during normal operation and anticipated operational occurrences (including refueling; purging; fuel handling and storage; radioactive material handling, processing, use, storage, and disposal; maintenance; routine operational surveillance; inservice inspection; and calibration). It should provide information on facility and equipment design, the planning and procedures programs, and the techniques and practices employed by the applicant in meeting the standards for protection against radiation of 10 CFR Part 20 and the guidance given in the appropriate regulatory guides, where the practices set forth in such guides will be used to implement NRC regulations. Reference to other chapters for information needed in this chapter should be specifically made where required.

12.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)

12.1.1 Policy Considerations

Describe the management policy and organizational structure related to ensuring that occupational radiation exposures are ALARA. Describe the applicable responsibilities and the related activities to be conducted by the management individuals having responsibility for radiation protection and the policy of maintaining occupational exposures ALARA. In the PSAR, describe policy with respect to designing and constructing the plant; in the FSAR, describe the ALARA policy as it will be applied to plant operations. In the PSAR, indicate whether, and if so how, the ALARA policy guidance given in Section C.1 of Regulatory Guide 8.8 (Ref. 1) and in Regulatory Guides 8.10 (Ref. 2) and 1.8 (Ref. 3) will be followed; if it will not be followed, describe the specific alternative approaches to be used. Indicate how the requirements of 10 CFR Part 20 (Ref. 4) will be met.

12.1.2 Design Considerations

In the PSAR, describe how experience from past designs and from operating plants is used to develop improved design for ensuring that occupational radiation exposures are ALARA. Describe how ALARA design guidance (both general and specific) is given to the individual designers. Describe how the design is directed toward reducing the need for maintenance of equipment and to reducing radiation levels and time spent where maintenance and other operational activities are required. Describe any mechanisms that provide for design review by a competent professional in radiation protection such as the utility radiation protection manager. These descriptions should be detailed in the PSAR, including an indication of whether, and if so how, the design consideration guidance provided in Section C.1 of Regulatory Guide 8.8 will be followed; if it will not be followed, describe the specific alternative approaches to be used.

The detailed facility design features for radiation protection and for ensuring that occupational radiation exposures will be ALARA should be covered in Section 12.3.1.

12.1.3 Operational Considerations

In the PSAR, describe the methods to be used to develop the detailed operational plans and procedures for ensuring that occupational radiation exposures are ALARA. Describe how these operational plans and procedures will impact on the design of the facility and how such planning has incorporated information from operating plant experience, other designs, etc. Describe how operational requirements are reflected in the design considerations described in Section 12.1.2 and the radiation protection design features described in Section 12.3.1. Indicate the extent to which the guidance on operational considerations given in Section C.1 of Regulatory Guide 8.8 and in Regulatory Guide 8.10 will be followed; if the guidance will not be followed, describe the specific alternative approaches to be used.

In the FSAR, provide the criteria and/or conditions under which various operating procedures and techniques for ensuring that occupational radiation exposures are ALARA are implemented for all systems that contain, collect, store, or transport radioactive liquids, gases, and solids (including, for example, the turbine system (for BWRs); the nuclear steam supply system; the residual heat removal systems; the spent fuel transfer, storage, and cleanup systems; and the radioactive waste treatment, handling, and storage systems). Describe means for planning and developing procedures for such radiation-exposure-related operations as maintenance, inservice inspections, radwaste handling, and refueling in a manner that will ensure that the exposures are ALARA. Describe any changes in operating procedures that result from the ALARA operational procedures review.

12.2 Radiation Sources

12.2.1 Contained Sources

In the PSAR, the sources of radiation that are the bases for the radiation protection design should be described in the manner needed as input to the shield design calculation. Those sources that are contained in equipment of the radioactive waste management systems should be described. In this section, source descriptions should be provided for other sources such as the reactor core, the spent fuel storage pool, various auxiliary systems, the steam lines and turbine system (including reheaters, moisture separators, etc.) as sources of N-16 in a BWR, and the equipment, systems, and piping containing activation product sources. For the reactor core, describe the source as it is used to determine radiation levels external to the biological shield at locations where occupancy may be required. For other sources, the description should tabulate sources by isotopic composition or gamma ray energy groups, strength (curie content), and geometry, as well as provide the basis for the values. The source location in the plant should be specified so that all important

143 230

sources of radioactivity can be located on plant layout drawings. For all the sources identified above, including activation product sources, the models and parameters for calculating the source magnitudes should be provided. Indicate whether, and if so how, the applicable guidance provided in ANSI N237 (Ref. 5) has been followed; if not followed, describe the specific alternative methods used. Describe any required byproduct, source, and special nuclear material (Refs. 6, 7, and 8) that may require shielding design considerations. In the FSAR, provide a listing of isotope, quantity, form, and use of all sources in this latter category that exceed 100 millicuries. Provide additional details (and any changes) of source descriptions that are used to develop the final shield design.

12.2.2 Airborne Radioactive Material Sources

In the PSAR, the sources of airborne radioactive material in equipment cubicles, corridors, and operating areas normally occupied by operating personnel should be described in the manner required for design of personnel protective measures and dose assessment. Those airborne radioactivity sources that have to be considered for their contribution to the plant effluent releases through the radioactive waste management system or the plant ventilation systems should be described in Chapter 11. Any other sources of airborne radioactivity in the areas mentioned above that are not covered in Chapter 11 should be included and described here. Sources resulting from reactor vessel head removal, relief valve venting, and movement of spent fuel should be included. The description should include a tabulation of the calculated concentrations of airborne radioactive material by nuclides expected during normal operation and anticipated operational occurrences for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. The models and parameters for calculating airborne radioactivity concentrations should be provided. In the FSAR, describe any changes or additions to the source data since the PSAR

12.3 Radiation Protection Design Features

12.3.1 Facility Design Features

In the PSAR, describe equipment and facility design features used for ensuring that occupational radiation exposures are ALARA. Indicate whether, and if so how, the design feature guidance given in Section C.2 of Regulatory Guide 8.8 has been followed; if not followed, describe the specific alternative approaches used.

Provide illustrative examples of the facility design features used in the PSAR design stage as applied to the systems listed in Section 12.1.3. The description should include those features that reduce need for maintenance and other operations in radiation fields, reduce radiation sources where operations must be performed, allow quick entry and easy access, provide remote operation capability, or reduce the time required for work in radiation fields and any other features that reduce radiation exposure of personnel. It should include descriptions of methods for

143 231

reducing the production, distribution, and retention of activation products through design methods, material selection, water chemistry, decontamination procedures, etc. An illustrative example should be provided for each of the following components (including equipment and piping layouts): liquid filters, demineralizers, absorber beds, particulate filters, recombiners, tanks, evaporators, pumps, steam generators, valve operating stations, and sampling stations. In the FSAR, the location of sampling ports, instrumentation, and control panels should be provided.

In the PSAR, provide scaled layout and arrangement drawings of the facility showing the locations of all sources described in Section 12 Provide on the layouts the radiation zone designations, including zone boundaries for both normal operational and refueling outage conditions. Reference other chapters as appropriate. The layouts should show shield wall thicknesses, controlled access areas, personnel and equipment decontamination areas, contamination control areas, traffic patterns, location of the health physics facilities, location* of airborne radioactivity and area radiation monitors, location of control panels for radwaste equipment and components, location of the onsite laboratory for analysis of chemical and radioactivity samples, and location of the counting room. Specify the design basis radiation level in the counting room during normal operation and anticipated operational occurrences. Describe the facilities and equipment such as hoods, glove boxes, filters, special handling equipment, and special shields that are related to the use of sealed and unsealed special nuclear, source, and byproduct material. In the FSAR, describe changes or additions to the radiation protection design since the PSAR.

12.3.2 Shielding

In the PSAR, provide information on the shielding for each of the radiation sources identified in Chapter 11 and Section 12.2, including the criteria for penetrations, the material, the method by which the shield parameters (cross sections, buildup factors, etc.) were determined, and the assumptions, codes, and techniques used in the calculations. Describe special protective features that use shielding, geometric arrangement (including equipment separation), or remote handling to ensure that occupational radiation exposures will be ALARA in normally occupied areas such as valve operating stations and sample collection stations. Indicate whether, and if so how, the guidance provided in Regulatory Guide 1.69 (Ref. 9) on concrete radiation shields and in Regulatory Guide 8.8 on special protective features has been followed; if not followed, describe the specific alternative methods used. In the FSAR, describe changes or additions in the shielding since the PSAR.

12.3.3 Ventilation

In the PSAR, the personnel protection features incorporated in the design of the ventilation system should be described. Those aspects of the design that relate to removing airborne radioactivity from equipment

*In the PSAR, if available, and update in the FSAR.

143 232

cubicles, corridors, and operating areas normally occupied by operating personnel and into the effluent control systems should be described in Chapter 11. Describe here any ventilation system protective features not covered in Chapter 11 or provided by the descriptions in Chapter 9. Include those aspects of the systems that relate to controlling the concentration of radioactivity in the areas mentioned above. Provide an illustrative example of the air cleaning system design, including an example layout of an air cleaning system housing showing filter mountings, access doors, aisle space, service galleries, and provisions for testing, isolation, and decontamination. Provide the criteria established for the change-out of air filters and adsorbers in the air cleaning system. Indicate whether, and if so how, the applicable guidance provided in Regulatory Guide 1.52 (Ref. 10) has been followed; if not followed, describe the specific alternative methods used. In the FSAR, include any changes or additions in the ventilation system design protective features since the PSAR.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

In the PSAR, describe the fixed area radiation and continuous airborne radioactivity monitoring instrumentation and the criteria for selection and placement.

In the FSAR, provide information on the auxiliary and/or emergency power supply and the range, sensitivity, accuracy, precision, calibration methods and frequency, alarm setpoints, recording devices, and location of detectors, readouts, and alarms for the monitoring instrumentation. Accident considerations and other needs for high range instrumentation should be included. In the FSAR, provide the location of airborne monitor sample collectors, and give details of sampling lines and pump location.

In the PSAR, describe the criteria and methods for obtaining representative in-plant airborne radioactivity concentrations, including airborne radioiodines and other radioactive materials, from the area being sampled.

In the FSAR, describe the radiation instrumentation that will be used to meet the criticality accident monitoring requirements of §70.24 of 10 CFR Part 70 for the storage area for new fuel.

Indicate whether, and if so how, the guidance provided by Regulatory Guides 1.21 (Ref. 11), 8.2 (Ref. 12), 8.8, 8.12 (Ref. 13), and 1.97 (Ref. 14) and ANSI N13.1-1969 (Ref. 15) has been followed; if not followed, describe the specific alternative methods used.

12.4 Dose Assessment

In the PSAR, provide the estimated annual occupancy (including numbers of personnel and durations of occupancy) of the plant radiation areas during normal operation and anticipated operational occurrences, including, for example, maintenance, repairs, replacement of pump and valves, and

143 233

plugging of steam generator tubes. For areas with expected airborne radioactivity concentrations (discussed in Section 12.2.2) during normal operation and anticipated operational occurrences as discussed above, provide estimated man-hours of occupancy and estimated inhalation exposures to personnel. Provide the objectives and criteria for design dose rates in various areas and an estimate of the annual man-rem doses associated with major functions such as operation, normal maintenance, radwaste handling, refueling, and inservice inspection. The basis, models, and assumptions for the above values should be provided. For routine or repetitive activities expected to occur with reasonably predictable frequencies and involving well-known sequences of operations, dose assessment should include, to the extent practicable, consideration of the specific plant and operation and should consider actual estimated dose rates at the various locations.

In the FSAR, provide updated estimates of annual man-rem doses for the functions listed above and the assumptions used in determining these values. Describe any changes made during planning or design review for the purpose of reducing these projected doses. Actual exposure data from similar operating plants operated in a similar manner may be used for the dose assessment for unpredictable activities but should be corrected for improvements in plant design and operating procedures.

In the PSAR, provide the estimated annual dose at the boundary of the restricted area (as defined in 10 CFR §20.3), at the site boundary, and, for multi-unit plants, at various locations in a new unit construction area from onsite radiation sources such as the turbine systems (for BWRs), the auxiliary building, the reactor building, and stored radioactive wastes and from radioactive effluents (direct radiation from the gaseous radioactive effluent plume). Provide estimated annual doses to construction workers due to radiation from these sources from the existing operating plant(s), and the annual man-rem doses associated with such construction. Include models, assumptions, and input data. In the FSAR, changes or additions since the PSAR should be provided. Indicate whether, and if so how, the guidance provided by Regulatory Guide 8.19 (Ref. 16) is followed; if not followed, describe the specific alternative methods used.

12.5 Health Physics Program

12.5.1 Organization

In the PSAR, describe the administrative organization of the health physics program, including the authority and responsibility of each position identified. Indicate whether, and if so how, the guidance of Regulatory Guides 8.2, 8.8, 8.10, and 1.8 has been followed; if not followed, describe the specific alternative approaches used. In the FSAR, describe the experience and qualification of the personnel responsible for the health physics program and for handling and monitoring radioactive materials, including special nuclear, source, and byproduct materials. Reference Chapter 13 as appropriate.

143 234

12.5.2 Equipment, Instrumentation, and Facilities

In the PSAR, provide the criteria for selection of portable and laboratory technical equipment and instrumentation for performing radiation and contamination surveys, for airborne radioactivity monitoring and sampling, for area radiation monitoring, and for personnel monitoring during normal operation, anticipated operational occurrences, and accident conditions. Describe the instrument storage, calibration, and maintenance facilities. Describe and identify the location of the health physics facilities (including locker rooms, shower rooms, offices, and access control stations), laboratory facilities for radioactivity analyses, protective clothing, respiratory protective equipment, decontamination facilities (for equipment and personnel), and other contamination control equipment and areas that will be available. Indicate whether, and if so how, the guidance provided by Regulatory Guides 8.3 (Ref. 17), 8.4 (Ref. 18), 8.8, 8.9 (Ref. 19), 8.12, 8.14 (Ref. 20), 8.15 (Ref. 21), and 1.97 has been followed; if not followed, describe the specific alternative methods used. In the FSAR, provide the location of the respiratory protective equipment, protective clothing, and portable and laboratory technical equipment and instrumentation. Describe the type of detectors and monitors and the quantity, sensitivity, range, and frequency and methods of calibration for all the technical equipment and instrumentation mentioned above.

12.5.3 Procedures

In the FSAR, the policy, methods, frequencies, and procedures for conducting radiation surveys should be described. Describe the procedures and methods of operation that have been developed for ensuring that occupational radiation exposures will be ALARA. Include a description of the procedures used in refueling, inservice inspections, radwaste handling, spent fuel handling, loading and shipping, normal operation, routine maintenance, and sampling and calibration that are specifically related to ensuring the radiation exposures will be ALARA. Describe the physical and administrative measures for controlling access and stay time for radiation areas. Reference may be made to Section 12.1, as appropriate. Describe the bases and methods for monitoring and control of contamination of personnel, equipment, and surface. Radiation protection training programs should be described. Indicate whether, and if so how, the guidance given in Regulatory Guides 8.2, 8.7 (Ref. 22), 8.8, 8.9, 8.10, 8.13 (Ref. 23), 1.8, 1.16 (Ref. 24), 1.33 (Ref. 25), and 1.39 (Ref. 26) will be followed; if it will not be followed, describe the specific alternative approaches to be used. Reference Chapter 13 as appropriate. Indicate how the requirements of 10 CFR Part 19 (Ref. 27) will be met.

Describe the methods and procedures for personnel monitoring (external and internal), including methods of recording, reporting, and analyzing results. Describe the program for internal radiation exposure assessment (whole body counting and bioassay), including the bases for selecting personnel who will be in the program, the frequency of their whole-body count and bioassay, and any nonroutine bioassay that will be performed.

Describe the methods and procedures for evaluating and controlling potential airborne radioactivity concentrations. Discuss any requirements for special air sampling and the issuance, selection, use, and maintenance of respiratory protective devices, including training programs and respiratory protective equipment fitting programs.

Method of handling and storage of sealed and unsealed byproduct, source, and special nuclear material should be described.

REFERENCES

1. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

2. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable."

3. Regulatory Guide 1.8, "Personnel Selection and Training."

4. 10 CFR Part 20, "Standards for Protection Against Radiation."

5. ANSI N237, "Source Term Specification," Final Draft, 1977.

6. 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material."

7. 10 CFR Part 40, "Domestic Licensing of Source Material."

8. 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material."

9. Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants."

10. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

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

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

13. Regulatory Guide 8.12, "Criticality Accident Alarm Systems."

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

143 236

15. ANSI N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities."

16. Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants--Design Stage Man-Rem Estimates."

17. Regulatory Guide 8.3, "Film Badge Performance Criteria."

18. Regulatory Guide 8.4, "Direct-Reading and Indirect-Reading Pocket Dosimeters."

19. Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program."

20. Regulatory Guide 8.14, "Personnel Neutron Dosimeters."

21. Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection."

22. Regulatory Guide 8.7, "Occupational Radiation Exposure Records Systems."

23. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

24. Regulatory Guide 1.16, "Reporting of Operating Information--Appendix A Technical Specifications."

25. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

26. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

27. 10 CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections."

143 237

13. CONDUCT OF OPERATIONS

This chapter of the SAR should provide information relating to the preparations and plans for operation of the plant. Its purpose is to provide assurance that the applicant will establish and maintain a staff of adequate size and technical competence and that operating plans to be followed by the licensee are adequate to protect public health and safety.

The information required at the PSAR stage pursuant to 10 CFR §50.34(a)(6), (9), and (10) should demonstrate adequate planning for the operational phase of the plant. The information required at the FSAR stage pursuant to 10 CFR §50.34(b)(6i), (6iv), (6v), and (7) should provide firm evidence that operating phase plans have been or are being implemented.

13.1 Organizational Structure Of Applicant

13.1.1 Management and Technical Support Organization

The description in this section of the corporate or home-office organization, its functions and responsibilities, and the number and the qualifications of personnel should be directed to activities that include facility design, design review, design approval, construction management, testing, and operation of the plant. The following specific information should be included.

13.1.1.1 Design and Operating Responsibilities. In the PSAR, the description should include the corporate functions and their specific responsibilities for the activities described in items 1 and 2 below and plans relative to item 3 below. In the FSAR, the description should summarize the degree to which the activities described in items 1 and 2 below have been accomplished, provide a schedule for completing these activities, and describe the specific responsibilities and activities relative to item 3 below.

1. Design and Construction Activities (Project Phase). The extent and assignment of these activities are generally contractual in nature and determined by the applicant. (Quality assurance aspects should be described in Section 17.1.) The following should be included:

a. Principal site-related engineering work such as meteorology, geology, seismology, hydrology, demography, and environmental effects,

b. Design of plant and ancillary systems,

c. Review and approval of plant design features,

d. Site layout with respect to environmental effects and security provisions,

e. Development of safety analysis reports,

f. Review and approval of material and component specifications,

g. Procurement of materials and equipment, and

h. Management and review of construction activities.

2. Preoperational Activities. These are the activities that should be substantially accomplished before preoperational testing begins and generally before submittal of the FSAR. The following should be included:

a. Development of human engineering design objectives and design phase review of proposed control room layouts,

b. Development and implementation of staff recruiting and training programs,

c. Development of plans for initial testing, and

d. Development of plant maintenance programs.

3. Technical Support for Operations. Technical services and backup support for the operating organization should become available prior to the initial testing program and continue throughout the life of the plant. The following are special capabilities that should be included:

a. Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgy and materials, and instrumentation and controls engineering,

b. Plant chemistry,

c. Health physics,

d. Fueling and refueling operations support, and

e. Maintenance support.

13.1.1.2 Organizational Arrangement. In the PSAR, the description should include organization charts reflecting the current headquarters and engineering structure and any planned modifications and additions to reflect the added functional responsibilities (described in 13.1.1.1) associated with the addition of the nuclear plant to the applicant's power generation capacity. The description should show how these responsibilities are delegated and assigned within and from the headquarters staff and the number of persons assigned or expected to be assigned to each of the working or performance level organizational units identified to implement these responsibilities.

In the FSAR, the description should include organization charts reflecting the current corporate structure and the specific working or performance level organizational units that will provide technical support for operation (Section 13.1.1.1, item 3). If these functions are to be

143 239

provided from outside the corporate structure, the contractual arrangements should be described.

13.1.1.3 Qualifications. The PSAR should describe general qualification requirements in terms of educational background and experience requirements for positions or classes of positions identified in 13.1.1.2. Personnel resumes should be provided for assigned persons identified in 13.1.1.2 holding key or supervisory positions in disciplines or job functions unique to the nuclear field or this project. For identified positions or classes of positions that have functional responsibilities for other than the identified application, the expected proportion of time assigned to the other activities should be described.

The FSAR should identify qualification requirements for headquarters staff personnel, which should be described in terms of educational background and experience requirements, for each identified position or class of positions providing headquarters technical support for operations. In addition, the FSAR should include resumes of individuals already employed by the applicant to fulfill responsibilities identified in item 3 of Section 13.1.1.1, including that individual whose job position corresponds most closely to that identified as "engineer in charge."

13.1.2 Operating Organization

This section of the SAR should describe the structure, functions, and responsibilities of the onsite organization established to operate and maintain the plant. The following specific information should be included.

13.1.2.1 Plant Organization. Provide an organization chart showing the title of each position, the number of persons assigned to common or duplicate positions (e.g., technicians, shift operators, repairmen), the number of operating shift crews, and the positions for which reactor operator and senior reactor operator licenses are required. For multi-unit stations, the organization chart (or additional charts) should clearly reflect planned changes and additions as new units are added to the station. The schedule, relative to the fuel loading date for each unit, for filling all positions should be provided.

13.1.2.2 Plant Personnel Responsibilities and Authorities. The functions, responsibilities, and authorities of plant positions corresponding to the following should be described:

1. Overall plant management,

2. Operations supervision,

3. Operating shift crew supervision,

4. Licensed operators,

143 240

5. Unlicensed operators,

6 Technical supervision,

7. Nuclear engineering supervision,

8. Radiation protection supervision,

9. Instrumentation and controls engineering supervision,

10. Instrumentation and controls maintenance supervision,

11. Equipment maintenance supervision, and

12. Quality assurance and quality control supervision.

For each position, where applicable, required interfaces with offsite personnel or positions identified in 13.1.1 should be described. Such interfaces include defined lines of reporting responsibilities, e.g., from the plant manager to his immediate supervisor, as well as functional or communication channels. In the FSAR, the following should also be described:

1. The line of succession of authority and responsibility for overall station operation through at least three persons, in the event of unexpected contingencies of a temporary nature, and

2. The delegation of authority to operating supervisors and to shift supervisors, including the authority to issue standing or special orders.

If the station contains, or is planned to contain, power generating facilities other than those relating to the application in question, this section should also describe interfaces with the organizations operating such other facilities. The description should include any proposed sharing of persons between the units and the proportion of their time that they will routinely and nonroutinely be assigned to the other unit.

13.1.2.3 Operating Shift Crews. The position titles, applicable operator licensing requirements for each, and the minimum numbers of personnel planned for each shift should be described for all combinations of units proposed to be at the station in either operating or cold shutdown mode. Also describe shift crew staffing plans unique to refueling operations. In addition, the proposed means of assigning shift responsibility for implementing the radiation protection program on a round-the-clock basis should be described.

13.1.3 Qualifications of Nuclear Plant Personnel

13.1.3.1 Qualification Requirements. This section of the SAR should describe the education, training, and experience requirements established for each management, operating, technical, and maintenance position category in the operating organization described in Section 13.1.2. Regulatory Guide 1.8, "Personnel Selection and Training," contains guidance on selection and training of personnel. The SAR should specifically indicate a commitment to meet the regulatory position stated in this guide or provide an acceptable alternative. Where a clear correlation cannot be made between the proposed plant staff positions and those referenced by Regulatory Guide 1.8, each position on the plant staff should be listed along with the corresponding position referenced by Regulatory Guide 1.8, or with a detailed description of the proposed qualifications for that position.

13.1.3.2 Qualifications of Plant Personnel (FSAR). The qualifications of the initial appointees to (or incumbents of) plant positions should be presented in resume format for key plant managerial and supervisory personnel through the shift supervisory level. The resumes should identify individuals by position title and, as a minimum, describe the individual's formal education, training, and experience (including any prior AEC or NRC licensing).

13.2 Training

13.2.1 Plant Staff Training Program

The PSAR should provide a description of the proposed training program in nuclear technology and other subjects important to safety for the entire plant staff. The FSAR should describe the training program as actually carried out up to the time of FSAR preparation and should note any significant changes from the program described in the PSAR. Regulatory Guide 1.8, "Personnel Selection and Training," provides guidance on an acceptable basis for relating initial training programs to plant staff positions. The PSAR and FSAR should indicate whether this guidance will be followed. If such guidance will not be followed, specific alternative methods that will be used should be described along with a justification for their use. A list of Commission regulations, guides, and reports pertaining to training of licensed and unlicensed nuclear power plant personnel is provided in Section 13.2.3.

13.2.1.1 Program Description. The program description should include the following information with respect to the formal training program in nuclear technology and other subjects important to safety (related technical training) for all plant management and supervisory personnel, Licensed Senior Operator (SRO) and Licensed Operator (RO) candidates, technicians, and general employees.

The PSAR should include:

1. The proposed subject matter of each course, the duration of the course (approximate number of weeks in full-time attendance), the

143 242

organization teaching the course or supervising instruction, and the position titles for which the course is given.

2. A description of proposed reactor operations experience training by nuclear power plant simulator or by assignment to a similar plant, including length of time (weeks), identity of simulator or plant, and identification by position of personnel to be trained.

3. A commitment to conduct an onsite formal training program and on-the-job training before initial fuel loading.

4. Any difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to §55.25 of 10 CFR Part 55 based on the extent of previous nuclear power plant experience. Experience groups should include the following:

a. Individuals with no previous experience,

b. Individuals who have had nuclear experience at facilities not subject to licensing,

c. Individuals who hold, or have held, licenses for comparable facilities.

5. A commitment to conduct an initial fire protection training program for the plant staff including:

a. Provisions for drills during construction, and

b. Provisions for indoctrination of construction personnel, as necessary.

The initial training should be completed prior to receipt of fuel at the site.

6. A detailed description of the training program for the individual(s) responsible for formulating and ensuring the implementation of the fire protection program. The training program should be consistent with the information on fire protection systems provided in Section 9.5.1.

7. Means for evaluating the training program effectiveness for all employees. For individuals seeking an operator license prior to criticality, this includes the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to paragraph 55.25(b) of 10 CFR Part 55.

The FSAR should include:

1. The proposed subject matter of each course, including a syllabus or equivalent course description, the duration of the course (approximate

number of weeks in full-time attendance), the organization teaching the course or supervising instruction, and the position titles for which the course is given.

2. A description of reactor operations experience training by nuclear power plant simulator or by assignment to a similar plant, including length of time (weeks), identity of simulator or plant, and identification by position of personnel to be trained.

3. The details of the onsite training program, including a syllabus or equivalent course description, the duration of the course (approximate number of weeks in full-time attendance), the organization teaching the course or supervising instruction, and the position title for which the course is given. The program should distinguish between classroom training and on-the-job training before and after the initial fuel loading.

4. Any difference in the training programs for individuals who will be seeking licenses prior to criticality pursuant to §55.25 of 10 CFR Part 55 based on the extent of previous nuclear power plant experience. Experience groups should include the following:

a. Individuals with no previous experience,

b. Individuals who have had nuclear experience at facilities not subject to licensing,

c. Individuals who hold, or have held, licenses for comparable facilities.

5. A detailed description of the fire protection training and retraining for the initial plant staff and replacement personnel. The program should describe:

a. The training planned for each member of the fire brigade,

b. The frequency of periodic firefighting drills,

c. The training provided for all remaining staff members, including personnel responsible for maintenance and inspection of fire protection equipment,

d. The indoctrination and training provided for persons temporarily assigned onsite duties during shutdown and maintenance outages, particularly those allowed unescorted access, and

e. The training provided for the fire protection staff members.

The description should include the course of instruction, the number of hours of each course, and the organization conducting the training.

143 244

6. Means for evaluating the training program effectiveness for each employee. For individuals seeking an operator license prior to criticality, this includes the means to be employed to certify that each applicant has had extensive actual operating experience pursuant to paragraph 55.25(b) of 10 CFR Part 55.

13.2.1.2 Coordination with Preoperational Tests and Fuel Loading. The PSAR should include a chart that shows the schedule of each part of the training program for each functional group of employees in the organization in relation to the schedule for preoperational testing, expected fuel loading, expected time for examinations prior to plant criticality for licensed operators, and expected time for examinations for licensed operators following plant criticality. In the FSAR, the applicant should include in the chart contingency plans for individuals applying for licenses prior to criticality in the event fuel loading is substantially delayed from the date indicated in the FSAR.

In the FSAR, the chart should reflect the extent to which the training program has been accomplished as of the approximate time of submittal of the FSAR.

13.2.2 Replacement and Retraining (FSAR)

This section should describe the applicant's plans for retraining of the plant staff, including requalification training for licensed operators and a commitment to provide training for replacement personnel.

13.2.2.1 Licensed Operators - Requalification Training. A detailed description of the applicant's licensed operator requalification training program should be provided. This description should show how the program will implement the requirements of Appendix A, "Requalification Programs for Licensed Operators of Production and Utilization Facilities," to 10 CFR Part 55.

13.2.2.2 Refresher Training for Unlicensed Personnel. The additional position categories on the plant staff for which retraining will be provided should be identified, and the nature, scope, and frequency of such retraining should be described.

13.2.2.3 Replacement Training. The applicant should briefly describe the training program for replacement personnel.

13.2.3 Applicable NRC Documents

The NRC regulations, regulatory guides, and reports listed below provide information pertaining to the training of nuclear power plant personnel. The SAR should indicate the extent to which the applicable portions of the guidance provided will be used and should justify any exceptions. Material discussed elsewhere in the SAR may be referenced.

143 245

1. 10 CFR Part 50, "Licensing of Production and Utilization Facilities."

2. 10 CFR Part 55, "Operators' Licenses."

3. 10 CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections."

4. Regulatory Guide 1.8, "Personnel Selection and Training."

5. Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants."

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

7. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

8. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable."

9. Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

10. "Utility Staffing and Training for Nuclear Power," WASH-1130, Revised June 1973.

11. NRC Operator Licensing Guide, NUREG-0094, July 1976.

13.3 Emergency Planning

This section of the SAR should describe the applicant's plans for coping with emergencies pursuant to paragraphs (a)(10) and (b)(6)(v) of §50.34 of 10 CFR Part 50. The items to be discussed are set forth in Appendix E, "Emergency Plans for Production and Utilization Facilities," to 10 CFR Part 50. Guidance is provided in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants." The information provided should also contribute to a determination that the exclusion area and the low population zone for the site comply with the definitions of paragraphs (a) and (b) of §100.3 of 10 CFR Part 100.

13.3.1 Preliminary Planning (PSAR)

At the PSAR stage, the items requiring description are set forth in the introductory paragraph and paragraphs A through G of Section II of Appendix E to 10 CFR Part 50. The following statements clarify information requirements applicable to the paragraphs indicated.

143 246

With respect to paragraph B, the PSAR should identify the agency with primary responsibility for emergency preparedness planning for situations involving real or potential radiological hazards in the State where the facility is to be located. The arrangements that the applicant has made with this agency for coordinating emergency response plans for the environs of the plant should be explained. Similar arrangements with the appropriate agency of any neighboring State should be described if any part of the neighboring State is within the low population zone (LPZ). Arrangements made or to be made to accommodate the radiological emergency preparedness responsibilities of neighboring States whose borders are beyond the LPZ but still within approximately 5 miles of the plant should also be described. Other State agencies that are expected to have emergency response roles should be identified. Local government organizations (agencies) having jurisdictional authority and responsibility for emergency response within the boundaries of the proposed LPZ should be identified. The PSAR should also identify the nearest Regional Coordinating Office for the U.S. Department of Energy's Radiological Assistance Program, and any Federal agency and appropriate regional or local office thereof, having a jurisdictional responsibility on lands or waterways within the boundaries of the proposed LPZ.

With respect to paragraph C, measures proposed to be taken to cope with the following emergency situations should be succinctly described:

a. Personnel injuries occurring at the plant site,

b. Fire emergencies,

c. Severe natural phenomena in the environment, and

d. Accidents that could lead to small, moderate, or substantial releases of radioactivity to the environment.

With respect to such releases, the discussion should, in particular, address accident assessment measures to be taken by personnel within the plant, with emphasis on the methodology to be employed for real-time estimates of the potential consequences (e.g., projected doses) and criteria for notifying appropriate offsite authorities who may need to implement protective actions. References may be made to other parts of the PSAR for relevant information.

The PSAR should also confirm that one of the protective measures that is to be incorporated in coordinated emergency plans is evacuation of persons from the exclusion area and from any other potentially affected sector of the environs extending at a minimum to the outer boundary of the proposed LPZ. An analysis, including specific information and findings that will be needed to ensure the development of adequately coordinated emergency plans with respect to evacuation as a protective measure, should be provided, including the following:

143 247

1. Plots showing projected ground-level doses for stationary individuals, for both whole body and thyroid, resulting from the most serious design basis accident analyzed in the SAR. These should be based on the same isotopic release rates to the atmosphere (most conservative case) as those used in Chapter 15 of the PSAR for the purpose of showing conformance to the guideline dose criteria of 10 CFR Part 100. Relative concentrations (χ over Q) for the first 2 hours should be the same as those used for establishing conformance to the siting criteria of 10 CFR Part 100. Dose contributions for subsequent time intervals and for distances beyond the exclusion area boundary should reflect reasonable time averaging of calculated relative concentrations, plume front transit times, and radioactive decay in transit. Dose conversion factors should be based on Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I;" if not, the bases selected should be described. The bases for developing the plots should be fully described. These data should be presented in the following format:

a. Use an appropriate scale with time (hours) following onset of release as the ordinate and distance (miles) from the release point as the abscissa. Sufficient background grid lines should be included to permit reasonable interpolation by eye.

b. Provide curves for whole body doses of 1, 5, and 25 rem and thyroid doses of 5, 25, 150, and 300 rem. Each curve should represent an estimate of the elapsed time to reach the specified dose level as a function of distance from the release point under the conditions postulated.

c. Extend each curve to an ordinate of not less than 8.0 hours either from an ordinate of 2.0 hours or from an abscissa equal to the exclusion radius, whichever results in the greater range of coverage. If any such curve does not intersect the outer LPZ boundary, it should be extended to such intersection or to an elapsed time of 24 hours, whichever occurs first.

2. The expected accident assessment time. This figure should incorporate the time required to identify and characterize this accident, the time needed to predict the projected doses resulting from the accident, and the time to notify offsite authorities. Include sufficient information to support the estimate. Reference should be made to other parts of the PSAR, as necessary, where the subject matter of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," is addressed.

3. An estimate of the minimum elapsed time that offsite authorities might require before an initial warning to the public can be given.

4. An estimate of the elapsed time from the first warning to a member of the public that may be required to warn all resident and transient persons within the potential evacuation areas determined in item 5

143 248

below. Discuss the means that might be employed to provide such warnings for both daytime and nighttime conditions.

5. An estimate of elapsed times, measured from the time of initial warning to persons, to evacuate (a) the exclusion area and (b) defined sectors of the environs. Sectors of the environs chosen for this analysis may be bounded by geographical or man-made features but should generally cover an arc of not less than 45° centered on the plant. They should extend outward at least to the outer boundary of the proposed LPZ.*

6. Information that should be provided in support of the estimates of item 5 above include:

a. A map showing all roads available for vehicular evacuation of the exclusion area and environs extending at least 10 miles from the plant. Road network information to be shown on or keyed to the map should include the character of each road, all intersections, the number of lanes, whether improved or unimproved, and other factors that may affect vehicular traffic capacities.

b. On the same or similar map, demographic data, both resident and transient, in 1-mile annular increments, out to the sector boundaries defined in item 5 above and for each such sector. Population levels projected as peak values during the expected life of the plant should be used. If this information is incorporated elsewhere in the SAR, a specific reference thereto is suitable.

c. If means other than the use of private automobiles are assumed for any of the evacuation time estimates of item 5 above, these should be specified.

7. The identity and locations of the agency or agencies that would be responsible for providing warning and direction to offsite persons.

With respect to paragraph E, the PSAR should identify offsite hospital facilities that are expected to be able to provide (a) emergency care and (b) definitive patient care for acute radiation injury. Evidence should be given that preliminary contact with a relatively nearby hospital has established a willingness and potential capability to receive and treat individuals from the plant site who may have been affected by radiological emergencies.

With respect to paragraph F, the required descriptions of training planned for employees may incorporate by reference other sections of the PSAR as applicable and appropriate, for example, in the areas of radiation protection and firefighting. The PSAR should also include commitments to provide site familiarity training for others, not employed at the site,

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*If the proposed LPZ boundary is less than 5 miles from the plant, prior consultation with the NRC staff is suggested before submitting the analysis requested in this section.

143 249

who may require such familiarity to discharge their responsibilities in an emergency and a description of the training that ensures qualified medical services in emergency situations.

Although applicable primarily to the FSAR stage, Annex A of Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," particularly Sections 5, 6, and 8.1.1, should be consulted for guidance at the PSAR stage.*

13.3.2 Emergency Plan (FSAR)

At the FSAR stage, a comprehensive emergency plan should be submitted. This plan should be a physically separate document identified as Section 13.3 of the FSAR. The plan should show how the objectives and requirements of Parts I and III and paragraphs A to J of Part IV of Appendix E to 10 CFR Part 50 are to be implemented. Regulatory Guide 1.101* should be consulted. The information requirements identified in paragraphs 1, 6.a, and 6.b above for the PSAR should be provided in an appendix to the emergency plan, including any changes that may be necessary to update information previously submitted.

13.4 Review and Audit

The SAR should describe the applicant's plans for conducting reviews and audits of operating phase activities that are important to safety. The primary focus of attention should be:

1. On the procedures that will implement the licensee's responsibility pursuant to §50.59 of 10 CFR Part 50 relating to proposed changes, tests, and experiments, and

2. On the procedures for after-the-fact review and evaluation of unplanned events. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," contains guidance on conducting reviews and audits.

The PSAR should specifically indicate a commitment to meet this guidance or describe alternative means for meeting the same objectives. The FSAR should describe the applicant's detailed plans for conducting reviews and audits of operating phase activities that are important to safety.

13.4.1 Onsite Review (FSAR)

This section should specifically describe how the onsite organization functions with respect:

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*The specific revision number and date of Regulatory Guide 1.101 used by the applicant should be specified.

143 250

1. To review of proposed changes to systems or procedures, tests, and experiments, and

2. To unplanned events that have operational safety significance.

The description should indicate how qualified members of the onsite operating organization will participate in the review of operating activities, either as part of their individual job responsibilities or as members of a functional review organization, to assist the plant manager.

13.4.2 Independent Review (FSAR)

This section should provide a detailed description of the provisions for performance of independent reviews of operating activities. Information in this section should describe the organizational method, composition and qualifications of the group, subjects to be reviewed, and the time such program is to be implemented relative to fuel loading of the (first) unit.

13.4.3 Audit Program (FSAR)

This section should provide a detailed description of the procedures and organization employed to implement the audit program with respect to operating activities and to verify compliance with the administrative controls and the quality assurance program.

13.5 Plant Procedures

This section of the SAR should describe administrative and operating procedures that will be used by the operating organization (plant staff) to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner. In general, the SAR is not expected to include detailed written procedures. The PSAR should provide preliminary schedules for their preparation, and the FSAR should provide a brief description of the nature and content of the procedures and a schedule for their preparation.

13.5.1 Administrative Procedures

13.5.1.1 Conformance with Regulatory Guide 1.33. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," contains guidance on facility administrative policies and procedures. The SAR should specifically indicate whether the applicable portions of Regulatory Guide 1.33 concerning plant procedures will be followed. If such guidance will not be followed, the SAR should describe specific alternative methods that will be used and the manner of implementing them.

13.5.1.2 Preparation of Procedures. The PSAR and FSAR should provide a schedule for the preparation of appropriate written administrative procedures (see Section 13.5.1.1). The FSAR should identify the persons (by

143 251

position) who have the responsibility for writing procedures and the persons who must approve them before they are implemented.

13.5.1.3 Procedures (FSAR). A description of administrative procedures should be provided and should include:

1. Standing orders to operations shift supervisors and shift crews including:

   a. The reactor operator's authority and responsibilities,

   b. The senior operator's authority and responsibilities,

   c. The responsibility to meet the requirements of 10 CFR §50.54(i), (j), (k), (l), and (m), including a diagram of the control area that indicates the area designated "at the controls."

2. Special orders of a transient or self-cancelling character.

3. Equipment control procedures.

4. Control of maintenance and modifications.

5. Master surveillance testing schedule.

6. Procedures for logbook usage and control.

7. Temporary procedures.

13.5.2 Operating and Maintenance Procedures (FSAR)

13.5.2.1 Control Room Operating Procedures. This section should describe primarily the procedures that are performed by licensed operators in the control room. Each such operating procedure should be identified by title and included in a described classification system. The general format and content for each class should be described. The following categories should be included, but need not necessarily form the basis for classifying these procedures:

1. System procedures.

2. General plant procedures.

3. Off-normal operating procedures.

4. Emergency procedures.

5. Alarm response procedures.

6. Temporary procedures.

143 252

In category 5, individual alarm response procedures should not be listed. However, the system employed to classify or subclassify alarm responses and the methods to be employed by operators to retrieve or refer to alarm response procedures should be described. Immediate action procedures required to be memorized should be identified.

13.5.2.2 Other Procedures. This section should describe how other operating and maintenance procedures are classified, what group or groups within the operating organization have the responsibility for following each class of procedures, and the general objectives and character of each class and subclass. The categories of procedures listed below should be included. If their general objectives and character are described elsewhere in the FSAR or the application, they may be described by specific reference thereto.

1. Plant radiation protection procedures.
2. Emergency preparedness procedures.
3. Instrument calibration and test procedures.
4. Chemical-radiochemical control procedures.
5. Radioactive waste management procedures
6. Maintenance and modification procedures.
7. Material control procedures.
8. Plant security procedures.

13.6 Industrial Security

This section of the SAR should note that the applicant's plans for physical protection of the facility are described in a separate part of the application withheld from public disclosure pursuant to §2.790(d), 10 CFR Part 2, "Rules of Practice." Detailed security measures for the physical protection of nuclear power plants are required by §50.34(c), of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," and applicable sections of 10 CFR Part 73, "Physical Protection of Plants and Materials." The regulatory position is set forth in Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage," and includes an endorsement of ANSI Standard N18.17-1973, "Industrial Security for Nuclear Power Plants."

13.6.1 Preliminary Planning (PSAR)

At the time of submittal of the PSAR, the applicant's separate submittal should describe plans for the screening of personnel who are to be employed to work at the proposed plant, including personnel selection policies, employee performance and evaluation procedures, and the industrial

143 253

security training program to be used to ensure that reliable and emotionally stable personnel are selected, maintained, and assigned to the plant staff and to the plant security force.

It should also describe plans for incorporating physical protection objectives and criteria into the design of the plant and the layout of equipment, including the following specific information as to how such plans will be or have been implemented:

1. Provide figures and/or drawings which identify the following:

a. Owner-controlled area, including private property markers, parking lot(s), and roads to be used for surveillance.

b. Protected area(s), including the associated isolation zone (clear area), physical barriers, access control points, lighting, intrusion monitoring and/or perimeter alarm systems, and roads or pathways to be used for surveillance.

c. Vital equipment and vital areas, including all access points.

d. Alarm station locations.

2. Describe the physical barrier construction for the protected and vital areas, and indicate the extent to which the positions set forth in ANSI N18.17-1973, Sections 3.3 and 3.4, are satisfied.

3. Describe the design features to be used for protecting all potential access points into the vital areas against unauthorized intrusion. Such features should include locking devices and intrusion detection devices.

4. Describe all intrusion alarms, emergency exit alarms, alarm systems, and line supervisory systems, and indicate the extent to which the level of performance and reliability specified by the Interim Federal Specification W-A-00450B (GSA-FSS), dated February 16, 1973, is met.

5. Describe the physical security provisions to be utilized in the design for the protection of security system service panels and wiring for protective devices, security communications systems, and door lock actuators.

6. Designate the person or group with the responsibility to conceive and detail security provisions in the physical plant design. If this responsibility is outside the owner organization, also specify the position within your organization responsible for the systematic review and control of the contracted activities.

143 254

13.6.2 Security Plan (FSAR)

At the time of submittal of the FSAR, the applicant's separate submittal should be a comprehensive description of the physical security program for the plant site. The information should include a description of the organization for security, a listing by title of all procedures to be established for plant security, access controls to the plant (including physical barriers and means of detecting unauthorized intrusions), provisions for monitoring the status of vital equipment, selection and training of personnel for security purposes, communication systems for security, provisions for maintenance and testing of security systems, and arrangements with law enforcement authorities for assistance in responding to security threats. The implementation schedule for the physical security program should be provided, including phases for multi-unit plants, where applicable.

Specific information for which guidance may be found in applicable referenced sections of ANSI N18.17-1973 and which should be included in the separate description is as follows:

1. Clear diagrams, to approximate scale, displaying the following:

a. Designated security areas of the plant site, including physical barriers,

b. The locations of alarm stations,

c. The locations of access control points to protected areas and vital areas,

d. The location of parking lots relative to the clear areas adjacent to the physical barriers surrounding protected areas,

e. Special features of the terrain that may present special vulnerability problems,

f. The location of relevant law enforcement agencies and their geographical jurisdictions.

2. If the policy of the owner organization permits use of any part of the owner-controlled area by members of the general public, describe in detail the extent to which the position of Section 3.2 of ANSI N18.17-1973 will be met.

3. The response capabilities of local law enforcement agencies should be fully described (Section 4.4 of ANSI N18.17-1973), including estimates of the number of officers that can arrive at the plant site, in the event of a security threat, within five to fifteen minutes, fifteen to thirty minutes, and thirty minutes to one hour after receipt of a call for assistance.

4. A description should be included of any provisions for alternative interim protective measures during periods when one or more components of the total security system are not functioning.

143 256

14. INITIAL TEST PROGRAM

This chapter of the Safety Analysis Report should provide information on the initial test program for structures, systems, components, and design features for both the nuclear portion of the plant and the balance of the plant. The information provided should address major phases of the test program, including preoperational tests, initial fuel loading and initial criticality, low-power tests, and power-ascension tests. The Preliminary Safety Analysis Report (PSAR) should describe the scope of the applicant's initial test program. The PSAR should also describe the applicant's general plans for accomplishing the test program in sufficient detail to show that due consideration has been given to matters that normally require advance planning. The Final Safety Analysis Report (FSAR) should describe the technical aspects of the initial test program in sufficient detail to show that the test program will adequately verify the functional requirements of plant structures, systems, and components and that the sequence of testing is such that the safety of the plant will not be dependent on untested structures, systems, or components. The FSAR should also describe measures which ensure that (1) the initial test program will be accomplished with adequate numbers of qualified personnel, (2) adequate administrative controls will be established to govern the initial test program, (3) the test program will be used, to the extent practicable, to train and familiarize the plant operating and technical staff in the operation of the facility, and (4) the adequacy of plant operating and emergency procedures will be verified, to the extent practicable, during the period of the initial test program.

14.1 Specific Information To Be Included In Preliminary Safety Analysis Reports

14.1.1 Scope of Test Program

The major phases of the initial test program should be described and the overall test objectives and general prerequisites for each major phase should be discussed.

The PSAR should describe how the initial test program will be applied to the nuclear portion as well as the balance-of-plant portion of the facility. The organizations, including those of the applicant, that will participate in the development and execution of the test program and the general responsibilities of these organizations should be described. The PSAR should describe the applicant's planned involvement in the development and approval of test procedures, conduct of the tests, and review and approval of test results. The applicant's plans for having responsible design organizations participate in establishing test performance requirements and acceptance criteria should be described along with the applicant's plans for contracting the work of planning, developing, or conducting portions of the test program. The method by which the applicant will retain responsibility for and maintain control of such contracted work should be discussed.

143 257

14.1.2 Plant Design Features That Are Special, Unique, or First of a Kind

A summary description of preoperational and/or startup testing planned for each unique or first-of-a-kind principal design feature should be included in the PSAR. The summary test descriptions should include the test method and test objectives.

14.1.3 Regulatory Guides

The PSAR should describe the applicant's plans for using guidance in applicable regulatory guides in the development and conduct of the initial test program. An example of such guidance is Regulatory Guide 1.68, "Initial Test Program for Water-Cooled Reactor Power Plants." If such guidance will not be followed, the PSAR should describe specific alternative methods along with a justification for their use.

14.1.4 Utilization of Plant Operating and Testing Experiences at Other Reactor Facilities

The PSAR should describe the applicant's plans for the utilization of available information on reactor plant operating experiences to establish where emphasis may be warranted in the test program. The schedule, relative to the fuel loading date, for conducting the study or implementing the program should be described.

14.1.5 Test Program Schedule

A summary description should be provided on the overall schedule, relative to the expected fuel loading date, for developing and conducting the major phases of the test program. Information provided should establish the scheduled time period for developing detailed test procedures and the scheduled time period for conducting the tests for each major phase. Information should be provided to establish the compatibility of the test program schedule with the schedules for hiring and training of the plant operating and technical staff and for development of plant operating and emergency procedures, or reference should be made to appropriate sections of Chapter 13 of the PSAR.

14.1.6 Trial Use of Plant Operating and Emergency Procedures

The applicant's plans pertaining to the trial use of plant operating and emergency procedures during the period of the initial test program should be described.

14.1.7 Augmenting Applicant's Staff During Test Program

The applicant's general plans for the assignments of additional personnel to supplement his plant operating and technical staff during each major phase of the test program should be described. The PSAR should provide a description of the general responsibilities of the various

143 258

augmenting organizations, a summary of the interrelationships and interfaces of the various organizations that will participate in the test program, the general qualifications of participating organizations, and the approximate schedule, relative to the fuel loading date, for augmenting the applicant's staff.

14.2 Specific Information To Be Included in Final Safety Analyis Reports

14.2.1 Summary of Test Program and Objectives

Describe the major phases of the test program and the specific objectives to be achieved for each major phase.

14.2.2 Organization and Staffing

A description of the applicant's organizational units and any augmenting organizations or other personnel that will manage, supervise, or execute any phase of the test program should be provided. This description should discuss the authorities, responsibilities, and degree of participation of each identified organizational unit and principal participants. The FSAR should describe how, and to what extent, the applicant's plant operating and technical staff will participate in each major test phase. Information pertaining to the experience and qualification of supervisory personnel and other principal participants that will be responsible for management, development, or conduct of each test phase should be provided or referenced elsewhere in the FSAR.

14.2.3 Test Procedures

The system that will be used to develop, review, and approve individual test procedures should be described, including the organizational units or personnel that are involved and their responsibilities. The FSAR should describe how organizations responsible for the design of the facility will participate in the establishment of performance requirements and acceptance criteria for testing plant structures, systems, and components and how such design organizations will interface with other participants involved in the test program. The FSAR should also describe the format of individual test procedures.

14.2.4 Conduct of Test Program

A description of the administrative controls that will govern the conduct of each major phase of the test programs should be provided. A description of the specific administrative controls that will be used to ensure that necessary prerequisites are satisfied for each major phase and for individual tests should also be provided. The FSAR should describe the methods to be followed in initiating plant modifications or maintenance that are determined to be required by the test program. The description should include the methods that will be used to ensure retesting following such modifications or maintenance and the involvement of

143 259

design organizations and the applicant in the review and approval of proposed plant modifications. The administrative controls pertaining to adherence to approved test procedures during the conduct of the test program and the methods for effecting changes to approved test procedures should be described.

14.2.5 Review, Evaluation, and Approval of Test Results

The measures to be established for the review, evaluation, and approval of test results for each major phase of the program should be described. The specific controls to be established to ensure notification of affected and responsible organizations or personnel when test acceptance criteria are not met and the controls established to resolve such matters should also be described. A discussion should be provided on the applicant's plans pertaining to (1) approval of test data for each major test phase before proceeding to the next test phase and (2) approval of test data at each power test plateau (during the power-ascension phase) before increasing power level.

14.2.6 Test Records

The applicant's requirements pertaining to the disposition of test procedures and test data following completion of the test program should be described.

14.2.7 Conformance of Test Programs with Regulatory Guides

The applicant should list all those regulatory guides applicable to initial test programs that he plans to use for his test program. If such guidance will not be followed, the FSAR should describe specific alternative methods along with justification for their use.

14.2.8 Utilization of Reactor Operating and Testing Experiences in Development of Test Program

Information on the applicant's program for utilizing available information on reactor operating experiences in the development of his initial test program should be described, including the status of the program. The organizations participating in the program should be identified, their roles in the program discussed, and a summary description of their qualifications provided. The sources and types of information reviewed, the conclusions or findings, and the effect of the program on the initial test program should be described.

14.2.9 Trial Use of Plant Operating and Emergency Procedures

The schedule for development of plant procedures should be provided as well as a description of how, and to what extent, the plant operating and emergency procedures will be use-tested during the initial test program.

14.2.10 Initial Fuel Loading and Initial Criticality

The FSAR should describe the procedures that will guide initial fuel loading and initial criticality, including the safety and precautionary measures to be established for safe operation.

14.2.11 Test Program Schedule

The schedule, relative to the fuel loading date, for conducting each major phase of the test program should be provided. If the schedule will overlap initial test program schedules for other reactors at the site, a discussion should be provided on the effects of such schedule overlaps on organizations and personnel participating in the initial test program. The sequential test schedule for testing individual plant structures, systems, and components should be provided. Each test required to be completed before initial fuel loading should be identified.

The schedule for the development of test procedures for each major phase of the initial test program, including the time that will be available for review by NRC field inspectors of approved procedures, prior to their use, should be discussed.

14.2.12 Individual Test Descriptions

Test abstracts for each individual test that will be conducted during the initial test program should be provided. Emphasis should be placed on system and design features that (1) are relied on for the safe shutdown and cooldown of the facility under normal and faulted conditions, (2) are relied on for establishing conformance with limits or limiting conditions for operation that will be established by the technical specifications, and (3) are relied on to prevent or to limit or mitigate the consequences of anticipated transients and postulated accidents. The abstracts should identify each test by title, specify the prerequisites and major plant operating conditions necessary for each test (such as power level and mode of operation of major control systems), provide a summary description of the test method, describe the test objectives, and provide a summary of the acceptance criteria for each test.

143 261

15. ACCIDENT ANALYSES

The evaluation of the safety of a nuclear power plant should include analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the Commission's construction permit and operating license reviews of plants.

In previous chapters of the SAR, the structures, systems, and components important to safety should have been evaluated for their susceptibility to malfunctions and failures. In this chapter, the effects of anticipated process disturbances and postulated component failures should be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

The situations analyzed should include anticipated operational occurrences (e.g., a loss of electrical load resulting from a line fault), off-design transients that induce fuel failures above those expected from normal operational occurrences, and postulated accidents of low probability (e.g., the sudden loss of integrity of a major component). The analyses should include an assessment of the consequences of an assumed fission product release that would result in potential hazards not exceeded by those from any accident considered credible.

Transient and Accident Classification

The approach outlined below is intended to organize the transients and accidents considered by the applicant and presented in the SAR in a manner that will:

1. Ensure that a sufficiently broad spectrum of initiating events has been considered,

2. Categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed, and

3. Permit the consistent application of specific acceptance criteria for each postulated initiating event.

To accomplish these goals, a number of disturbances of process variables and malfunctions or failures of equipment should be postulated. Each postulated initiating event should be assigned to one of the following categories:

1. Increase in heat removal by the secondary system (turbine plant),

 143 262

2. Decrease in heat removal by the secondary system (turbine plant),

3. Decrease in reactor coolant system flow rate,

4. Reactivity and power distribution anomalies,

5. Increase in reactor coolant inventory,

6. Decrease in reactor coolant inventory,

7. Radioactive release from a subsystem or component, or

8. Anticipated transients without scram.

Typical initiating events that are representative of those that should be considered by the applicant in this chapter of the SAR are presented in Table 15-1. The evaluation of each initiating event should be presented in a separate subsection corresponding to the eight categories defined above. The information to be presented in these subsections is outlined in Section 15.X.X.

One of the items of information that should be discussed for each initiating event relates to its expected frequency of occurrence. Each initiating event within the eight major groups should be assigned to one of the following frequency groups:

1. Incidents of moderate frequency,

2. Infrequent incidents, or

3. Limiting faults.

The initiating events for each combination of category and frequency group should be evaluated to identify the events that would be limiting. The intent is to reduce the number of initiating events that need to be quantitatively analyzed. That is, not every postulated initiating event needs to be completely analyzed by the applicant. In some cases a qualitative comparison of similar initiating events may be sufficient to identify the specific initiating event that leads to the most limiting consequences. Only that initiating event should then be analyzed in detail.

It should be noted, however, that different initiating events in the same category/frequency group may be limiting when the multiplicity of consequences are considered. For example, within a given category/frequency group combination, one initiating event might result in the highest reactor coolant pressure boundary (RCPB) pressure while another initiating event might lead to minimum core thermal-hydraulic margins or maximum offsite doses.

143 263

Plant Characteristics Considered in the Safety Evaluation

A summary of plant parameters considered in the safety evaluation should be given; e.g., core power, core inlet temperature, reactor system pressure, core flow, axial and radial power distribution, fuel and moderator temperature coefficient, void coefficient, reactor kinetics parameters, available shutdown rod worth and control rod insertion characteristics. A range of values should be specified for plant parameters that vary with fuel exposure or core reload. The range should be sufficiently broad to cover all expected changes predicted for the entire life of the plant. The permitted operating band (permitted fluctuations in a given parameter and associated uncertainties) on reactor system parameters should be specified. The most adverse conditions within the operating band should be used as initial conditions for transient analysis.

Assumed Protection System Actions

Settings of all protection system functions that are used in the safety evaluation should be listed. Typical protection system functions are reactor trips, isolation valve closures, ECCS initiation, etc. The uncertainty (combined effect of calibration error, drift, instrument error, etc.) associated with each function should also be listed together with the expected and maximum delay times.

15.X Evaluation of Individual Initiating Events

The applicant should provide an evaluation of each initiating event using the format of Section 15.X.X (e.g., 15.2.7 for a loss of normal feedwater flow initiating event). As shown in Table 15-1, a particular initiating event may be applicable to more than one category. The SAR sections should be appropriately referenced to indicate this.

The detailed information listed in Section 15.X.X, paragraphs 1 and 2, should be given for each initiating event. However, the extent of the quantitiative information in Section 15.X.X, paragraphs 3 through 5, that should be included will differ for the various initiating events. For those situations where a particular initiating event is not limiting, only the qualitative reasoning that led to that conclusion need be presented, along with a reference to the section that presents the evaluation of the more limiting initiating event. Further, for those initiating events that require a quantitative analysis, such an analysis may not be necessary for each of Section 15.X.X, paragraphs 3 through 5. For example, there are a number of plant transient initiating events that result in minimal radiological consequences. The applicant should merely present a qualitative evaluation to show this to be the case. A detailed evaluation of the radiological consequences need not be performed for each such initiating event.

143 264

15.X.X Event Evaluation

1. Identification of causes and frequency classification. For each event evaluated, include a description of the occurrences that lead to the initiating event under consideration. The probability of the initiating event should be estimated and the initiating event should be assigned to one of the following groups:

a. Incidents of moderate frequency - these are incidents, any one of which may occur during a calendar year for a particular plant.

b. Infrequent incidents - these are incidents, any one of which may occur during the lifetime of a particular plant.

c. Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

2. Sequence of events and systems operation. The following should be discussed for each initiating event:

a. The step-by-step sequence of events from event initiation to the final stabilized condition. This listing should identify each significant occurrence on a time scale, e.g., flux monitor trip, insertion of control rods begin, primary coolant pressure reaches safety valve set point, safety valves open, safety valves close, containment isolation signal initiated, and containment isolated. All required operator actions should also be identified.

b. The extent to which normally operating plant instrumentation and controls are assumed to function.

c. The extent to which plant and reactor protection systems are required to function.

d. The credit taken for the functioning of normally operating plant systems.

e. The operation of engineered safety systems that is required.

The effect of single failures in each of the above areas and the effect of operator errors should be discussed and evaluated. The discussion should provide enough detail to permit an independent evaluation of the adequacy of the system as related to the event under study. One method of systematically investigating single failures is the use of a plant operational analysis or a failure mode and effects analysis. The results of these types of analyses can be used to demonstrate that the safety actions required to mitigate the consequences of an event are provided by the safety systems essential to performing each safety action. A sample

143 265

format is described in Transactions of the American Nuclear Society 1973 Winter Meeting (November 11-15, 1973, pp. 339-340).

3. Core and system performance.

a. Mathematical model. The mathematical model employed, including any simplifications or approximations introduced to perform the analyses, should be discussed. Any digital computer programs or analog simulations used in the analyses should be identified. If a set of codes is used, the method combining these codes should be described. Important output of each code should be presented and discussed under "results." Principal emphasis should be placed on the input data and the extent or range of variables investigated. This information should include figures showing the analytical model, flow path identification, actual computer listing, and complete listing of input data. The detailed description of mathematical models and digital computer programs or listings are preferably included by reference to documents available to the NRC with only summaries provided in the SAR text.

b. Input parameters and initial conditions. The input parameters and initial conditions used in the analyses should be clearly identified. Table 15-2 provides a representative list of these items. However, the initial values of other variables and additional parameters should be included in the SAR if they are used in the analyses of the particular event being analyzed.

The parameters and initial conditions used in the analyses should be suitably conservative for the event being evaluated except that anticipated transient without scram (ATWS) analyses should use realistic initial values. The bases used to select the numerical values that are input parameters to the analysis, including the degree of conservatism, should be discussed in the SAR.

c. Results. The results of the analyses should be presented and described in detail in the SAR. Key parameters should be presented as a function of time during the course of the transient or accident. The following are examples of parameters that should be included:

(1) Neutron power,

(2) Thermal power,

(3) Heat fluxes, average and maximum,

(4) Reactor coolant system pressure,

(5) Minimum CHFR, DNBR, or CPR, as applicable,

(6) Core and recirculation loop coolant flow rates (BWRs),

(7) Coolant conditions - inlet temperature, core average temperature (PWR), core average steam volume fraction (BWR), average exit and hot channel exit temperatures, and steam volume fractions,

(8) Temperatures - maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy,

(9) Reactor coolant inventory - total inventory and coolant level in various locations in the reactor coolant system,

(10) Secondary (power conversion) system parameters - steam flow rate, steam pressure and temperature, feedwater flow rate, feedwater temperature, steam generator inventory, and

(11) ECCS flow rates and pressure differentials across the core, as applicable.

The discussion of results should emphasize the margins between the predicted values of various core parameters and the values of these parameters that would represent minimum acceptable conditions.

4. Barrier performance. This section of the SAR should discuss the evaluation of the parameters that may affect the performance of the barriers, other than fuel cladding, that restrict or limit the transport of radioactive material from the fuel to the public.

a. Mathematical model. The mathematical model employed, including any simplifications or approximations introduced to perform the analyses, should be discussed. If the model is identical, or nearly identical, with that used to evaluate core performance, this should be stated in the SAR. In that case, only the differences, if any, between the models need be described.

A detailed description of the model used to evaluate barrier performance should be presented if it is significantly different from the core performance model. The information that should be included is indicated in paragraph 3 of Section 15.X.X, item a.

b. Input parameters and initial conditions. Any input parameters and initial conditions of variables relevant to the evaluation of barrier performance that were not presented and discussed in paragraph 3 of Section 15.X.X., item b, should be discussed in this section. The discussion should present the numerical values of the input to the analyses and should discuss the degree of conservatism of the selected values.

c. Results. The results of the analyses should be presented and described in detail in the SAR. As a minimum, the following information should be presented as a function of time during the course of the transient or accident:

143 267

(1) Reactor coolant system pressure,

(2) Steam line pressure,

(3) Containment pressure,

(4) Relief and/or safety valve flow rate,

(5) Flow rate from the reactor coolant system to the containment system, if applicable.

5. Radiological consequences. This section of the SAR should summarize the assumptions, parameters, and calculational methods used to determine the doses that result from limiting faults and infrequent incidents. Sufficient information should be given in this section to fully substantiate the results and to allow an independent analysis to be performed by the NRC staff. Thus, this section should include all of the pertinent plant parameters that are required to calculate doses for the exclusion boundary and the low population zone as well as those locations within the exclusion boundary where significant site-related activities may occur (e.g., the control room).

The elements of the dose analysis that are applicable to several accident types or that are used many times throughout Chapter 15 can be summarized in this section (or cross-referred) with the bulk of the information appearing in appendices.

If there are no radiological consequences associated with a given initiating event, this section for the event should simply contain a statement indicating that containment of the activity was maintained and by what margin.

An analysis should be provided for each limiting event. These analyses should be based on design basis assumptions acceptable to the NRC for purposes of determining adequacy of the plant design to meet 10 CFR Part 100 criteria. These design basis assumptions can, for the most part, be found in regulatory guides that deal with radiological releases. For instance, when calculating the radiological consequences of a loss-of-coolant accident (LOCA), it is suggested that the assumptions given in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," be used. This analysis should be referred to as the "design basis analysis." There may be instances in which the applicant will not agree with the conservative margins inherent in the design basis approach approved by the NRC staff or the applicant may desire to provide a "realistic analysis" for comparison purposes. If this is the case, the applicant may provide an indication of the assumptions he believes to be adequately conservative, but the known NRC assumptions should nevertheless be used in the design basis analysis.

Any "realistic analysis" provided will help quantify the margins that are inherent in the design basis approach. A "realistic analysis" need not include a consequence assessment and may be limited to a presentation of assumptions that are more likely to be obtained than those used for purposes of design.

The parameters and assumptions used for these analyses, as well as the results, should be presented in tabular form. Table 15-3 provides a representative list of these items. Table 15-4 summarizes additional items that should be provided when dealing with specific types of accidents. When possible, the summary tabulation should provide the necessary quantitative information. If, however, a particular assumption cannot be simply or clearly stated in the table, the table should reference a section or an appendix that adequately discusses the information.

Judgment should be used in eliminating unnecessary parameters from the summary table or in adding parameters of significance that do not appear in Table 15-3 or 15-4. The summary table should have two columns. One column should indicate the assumptions used in the design basis analysis, while the other should indicate assumptions used in the realistic analysis.

A diagram of the dose computation model, labeled "Containment Leakage Dose Model," should be appended to Chapter 15. An explanation of the model should accompany the diagram. The purpose of the appendix is to clearly illustrate the containment modeling, the leakage or transport of radioactivity from one compartment to another or to the environment, and the presence of engineered safety features (ESF) such as filters or sprays that are called on to mitigate the consequences of the LOCA. The diagram should employ easily identifiable symbols, e.g., squares to represent the containment or various portions of it, lines with arrowheads drawn from one compartment to another or to the environment to indicate leakage or transport of radioactivity, and other suitably labeled or defined symbols to indicate the presence of ESF filters or sprays. Individual sketches (or equivalent) may be used for each significant time interval in the containment leakage history (e.g., separate sketches showing the pulldown of a dual containment annulus and the exhaust and recirculation phases once negative pressure in the annulus is achieved, with the appropriate time intervals given).

In presenting the assumptions and methodology used in determining the radiological consequences, care should be taken to ensure that analyses are adequately supported with backup information, either by reporting the information where appropriate, by referencing other sections within the SAR, or by referencing documents readily available to the NRC staff. Such information should include the following:

a. A description of the mathematical or physical model employed, including any simplifications or approximations introduced to perform the analyses.

143 269

b. An identification and description of any digital computer program or analog simulation used in the analysis. The detailed description of mathematical models and programs are preferably included by reference with only summaries provided in the SAR text.

c. An identification of the time-dependent characteristics, activity, and release rate of the fission products or other transmissible radioactive materials within the containment system that could escape to the environment via leakages in the containment boundaries and leakage through lines that could exhaust to the environment.

d. The considerations of uncertainties in calculational methods, equipment performance, instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

e. A discussion of the extent of system interdependency (containment system and other engineered safety features) contributing directly or indirectly to controlling or limiting leakages from the containment system or other sources (e.g., from spent fuel handling areas), such as the contribution of (1) containment water spray systems, (2) containment air cooling systems, (3) air purification and cleanup systems, (4) reactor core spray or safety injection systems, (5) postaccident heat removal systems, and (6) main steam line isolation valve leakage control systems (BWR).

This section should present the results of the dose calculations giving the potential 2-hour integrated whole body and thyroid doses for the exclusion boundary. Similarly, it should provide the doses for the course of the accident at the closest boundary of the low population zone (LPZ) and, when significant, the doses to the control room operators during the course of the accident. Other organ doses should be presented for those cases where a release of solid fission products or transuranic elements are postulated to be released to the containment atmosphere.

143 270

TABLE 15-1

REPRESENTATIVE INITIATING EVENTS
TO BE ANALYZED IN SECTIONS 15.X.X OF THE SAR

1. Increase in Heat Removal by the Secondary System

   1.1 Feedwater system malfunctions that result in a decrease in feedwater temperature

   1.2 Feedwater system malfunctions that result in an increase in feedwater flow

   1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow

   1.4 Inadvertent opening of a steam generator relief or safety valve

   1.5 Spectrum of steam system piping failures inside and outside of containment in a PWR

2. Decrease in Heat Removal by the Secondary System

   2.1 Steam pressure regulator malfunction or failure that results in decreasing steam flow

   2.2 Loss of external electric load

   2.3 Turbine trip (stop valve closure)

   2.4 Inadvertent closure of main steam isolation valves

   2.5 Loss of condenser vacuum

   2.6 Coincident loss of onsite and external (offsite) a.c. power to the station

   2.7 Loss of normal feedwater flow

   2.8 Feedwater piping break

3. Decrease in Reactor Coolant System Flow Rate

   3.1 Single and multiple reactor coolant pump trips

   3.2 BWR recirculation loop controller malfunctions that result in decreasing flow rate

   3.3 Reactor coolant pump shaft seizure

143 271

TABLE 15-1 (Continued)

3.4 Reactor coolant pump shaft break

4. Reactivity and Power Distribution Anomalies

4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling

4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power)

4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods

4.4 Startup of an inactive reactor coolant loop or recirculating loop at an incorrect temperature

4.5 A malfunction or failure of the flow controller in a BWR loop that results in an increased reactor coolant flow rate

4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR

4.7 Inadvertent loading and operation of a fuel assembly in an improper position

4.8 Spectrum of rod ejection accidents in a PWR

4.9 Spectrum of rod drop accidents in a BWR

5. Increase in Reactor Coolant Inventory

5.1 Inadvertent operation of ECCS during power operation

5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory

5.3 A number of BWR transients, including items 2.1 through 2.6 and item 1.2.

6. Decrease in Reactor Coolant Inventory

6.1 Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR

TABLE 15-1 (Continued)

6.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment

6.3 Steam generator tube failure

6.4 Spectrum of BWR steam system piping failures outside of containment

6.5 Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of containment in a BWR

6.6 A number of BWR transients, including items 2.7, 2.8, and 1.3

7. Radioactive Release from a Subsystem or Component

7.1 Radioactive gas waste system leak or failure

7.2 Radioactive liquid waste system leak or failure

7.3 Postulated radioactive releases due to liquid tank failures

7.4 Design basis fuel handling accidents in the containment and spent fuel storage buildings

7.5 Spent fuel cask drop accidents

8. Anticipated Transients Without Scram

8.1 Inadvertent control rod withdrawal

8.2 Loss of feedwater

8.3 Loss of a.c. power

8.4 Loss of electrical load

8.5 Loss of condenser vacuum

8.6 Turbine trip

8.7 Closure of main steam line isolation valves

143 273

TABLE 15-2

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS AND ACCIDENTS

Neutron Power

Moderator Temperature Coefficient of Reactivity

Moderator Void Coefficient of Reactivity

Doppler Coefficient of Reactivity

Effective Neutron Lifetime

Delayed Neutron Fraction

Average Heat Flux

Maximum Heat Flux

Minimum DNBR, CHFR, or CPR

Axial Power Distribution

Radial Power Distribution

Core Coolant Flow Rate

Recirculation Loop Flow Rate (BWR)

Core Coolant Inlet Temperature

Core Average Coolant Temperature (PWR)

Core Average Steam Volume Fraction (BWR)

Core Coolant Average Exit Temperature, Steam Quality, and Steam Void Fraction

Hot Channel Coolant Exit Temperature, Steam Quality, and Steam Void Fraction

Maximum Fuel Centerline Temperature

Reactor Coolant System Inventory (lb)

Coolant Level in Reactor Vessel (BWR)

Coolant Level in Pressurizer (PWR)

143 274

TABLE 15-2 (Continued)

Reactor Coolant Pressure

Steam Flow Rate

Steam Pressure

Steam Quality (temperature if superheated)

Feedwater Flow Rate

Feedwater Temperature

CVCS Flow and Boron Concentration (if these vary during the course of the transient or accident being analyzed)

Control Rod Worth, Differential, and Total

143 275

TABLE 15-3

REPRESENTATIVE PARAMETERS TO BE TABULATED*
FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Assumptions
1. Data and assumptions used to estimate radioactive source from postulated accidents		
a. Stretch power level		
b. Burnup		
c. Percent of fuel perforated		
d. Release of activity by nuclide		
e. Iodine fractions (organic, elemental, and particulate)		
f. Reactor coolant activity before the accident (and secondary coolant activity for PWR). Two values for primary system iodine activity concentration should be given. These two values should indicate (1) the maximum allowable equilibrium iodine concentration and (2) the maximum allowable concentration resulting from a preaccident iodine spike.		
2. Data and assumptions used to estimate activity released		
a. Primary containment volume and leak rate		
b. Secondary containment volume and leak rate		
c. Valve movement times		
d. Adsorption and filtration efficiencies		
e. Recirculation system parameters (flow rates versus time, mixing factor, etc.)		
f. Containment spray first order removal lambdas as determined in Section 6.2.3		
g. Containment volumes		
h. All other pertinent data and assumptions		

143 276

*As applicable to the event being described.

TABLE 15-3 (Continued)

	Design Basis Assumptions	Realistic Assumptions
3. Dispersion Data		
a. Location of points of release		
b. Distances to applicable receptors (e.g., control room, exclusion boundary, and LPZ)		
c. χ/Qs at control room, exclusion boundary, and LPZ (for time intervals of 2 hours, 8 hours, 24 hours, 4 days, 30 days)		
4. Dose Data		
a. Method of dose calculation		
b. Dose conversion assumptions		
c. Peak [or f(t)] concentrations in containment		
d. Doses (whole body and thyroid doses for LPZ and exclusion boundary; beta, gamma, and thyroid doses for the control room)		

143 277

TABLE 15-4

ADDITIONAL PARAMETERS AND INFORMATION TO BE PROVIDED OR REFERENCED IN THE SUMMARY TABULATION FOR SPECIFIC DESIGN BASIS ACCIDENTS

1. Loss-of-Coolant Accident (Section 15.6.5)

   a. Hydrogen Purge Analysis

      (1) Holdup time prior to purge initiation (assuming recombiners are inoperative)

      (2) Iodine reduction factor

      (3) χ/Q values at appropriate time of release

      (4) Purge rates for at least 30 days after initiation of purge

      (5) LOCA plus purge dose at LPZ

   b. Equipment Leakage Contribution to LOCA Dose

      (1) Iodine concentration in sump water after LOCA

      (2) Maximum operational leak rate through pump seals, flanges, valves, etc.

      (3) Maximum leakage assuming failure and subsequent isolation of a component seal

      (4) Total leakage quantities for (2) and (3)

      (5) Temperature of sump water vs time

      (6) Time intervals for automatic and operator action

      (7) Leak paths from point of seal or valve leakage to the environment

      (8) Iodine partition factor for sump water vs temperature of water

      (9) Charcoal adsorber efficiency assumed for iodine removal

   c. Main Steam Line Isolation Valve Leakage Control System Contribution to LOCA Dose (BWR)

      (1) Time of system actuation

      (2) Fraction of isolation valve leakage from each release point

TABLE 15-4 (Continued)

(3) Flow rates vs time for each release path

(4) Location of each release point

(5) Transport time to each release point

2. Waste Gas System Failure (Section 15.7.1)

a. Activity transfer times to waste gas system components

b. Number of tanks or other holdup components

c. Tank volumes

d. Charcoal bed delay times for Xe and Kr

e. Seismic classification of tank and associated piping

f. Decontamination factors of components

g. Primary coolant volume

h. Isotopic activity in each system component including daughter products

i. Time to isolate air ejector

j. Delay time in delay pipe

k. Design basis activity measured at air ejector (Ci/sec) including contribution due to activity spiking in coolant

3. Main Steam Line and Steam Generator Tube Failures (Sections 15.1.5, 15.6.3, 15.6.4)

a. Characterization of primary and secondary (PWR) system. Sufficient information should be given, as appropriate, to adequately describe the time-histories from accident initiation until accident recovery is complete for temperatures, pressures, steam generator water capacity, steaming rates, feedwater rates, blowdown rates, and primary-to-secondary leakage rates.

b. Potential increase in iodine release rate above the equilibrium value (i.e., iodine spiking) from the fuel to the primary coolant as a result of the accident or a pre-accident primary system transient.

c. Chronological list of system response times, operator actions, valve closure times, etc.

143 279

TABLE 15-4 (Continued)

d. Steam and water release quantities and all assumptions made in their computation

e. Description of the iodine transport mechanism and release paths between the primary system and the environment. The bases for an assumed partitioning of iodine between liquid and steam phases should be described and justified.

f. Possible fuel rod failure resulting from the accident, assuming the most reactive control rod remains in its fully withdrawn position.

g. Possible steam generator tube failure resulting from a PWR steam line break accident.

4. Fuel Handling Accident (in the Containment and Spent Fuel Storage Buildings) (Section 15.7.4)

a. Number of fuel rods in core

b. Number, burnup, and decay time of fuel rods assumed to be damaged in the accident

c. Radial peaking factor for the rods assumed to be damaged

d. Earliest time after shutdown that fuel handling begins

e. Amounts of iodines and noble gases released into pool

f. Pool decontamination factors

g. Time required to automatically switch from normal containment purge operation to either safety-grade filters or isolation

h. Amount of radioactive release not routed through ESF-grade filters.

The following items should be provided to determine if the calculational methods of Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," apply:

i. Maximum fuel rod pressurization

j. Minimum water depth between top of fuel rods and fuel pool surface

143 280

TABLE 15-4 (Continued)

k. Peak linear power density for the highest power assembly discharged

l. Maximum centerline operating fuel temperature for the fuel assembly in item k above

m. Average burnup for the peak assembly in item k above

5. Control Rod Ejection and Control Rod Drop Accidents (Sections 15.4.8 and 15.4.9)

   a. Percent of fuel rods undergoing clad failure

   b. Radial peaking factors for rods undergoing clad failure

   c. Percent of fuel reaching or exceeding melting temperature

   d. Peaking factors for fuel reaching or exceeding melting temperature

   e. Percent of core fission products assumed released into reactor coolant

   f. Summary of primary and secondary system parameters used to determine the activity release through the secondary system (PWRs only). The information specified in items 3a, b, c, d, and e of this table should also be provided for this accident.

   g. Summary of containment system parameters used to determine activity release terms from containment leak paths

   h. Summary of system parameters and decontamination factors used to determine activity release from condenser leak paths (BWR)

6. Spent Fuel Cask Drop (Section 15.7.5)

   a. Number of fuel elements in largest capacity cask

   b. Number, burnup, and decay time of fuel elements in cask assumed to be damaged

   c. Number, burnup, and decay time of fuel elements in pool assumed to be damaged as a consequence of a cask drop (if any)

   d. Average radial peaking factor for the rods assumed to be damaged

TABLE 15-4 (Continued)

e. Earliest time after reactor fueling that cask loading operations begin

f. Amounts of iodines and noble gases released into air and into pool

g. Pool decontamination factors, if applicable

7. Failure of Small Lines Carrying Primary Coolant Outside Containment (Section 15.6.2)

   a. Detailed description of primary system response, leakage rate, operator action, valve closure times, etc. Also, figures indicating primary system pressure and temperature and primary coolant leakage versus time for the duration of the accident should be provided, as well as chronological listing of system response times, operator actions, valve closure times, etc.

   b. Summary of primary system iodine activity during the accident and its effect on the calculated accident consequences as described in item 3.b of this table

   c. Iodine transport mechanism and release paths from the leak point to the environment

143 282

16. TECHNICAL SPECIFICATIONS

Section 50.36 of 10 CFR Part 50 requires that each operating license issued by the Commission contain Technical Specifications that set forth the limits, operating conditions, and other requirements imposed on facility operation for, among other purposes, the protection of the health and safety of the public. Each applicant for an operating license is required to submit proposed Technical Specifications and their bases for the facility. They should be consistent with the content and format of the Standard Technical Specifications available from the Commission for the appropriate nuclear steam supply system (NSSS) vendor. After review and needed modification by the NRC staff, these Technical Specifications will be issued by the Commission as Appendix A to the operating license.

16.1 Preliminary Technical Specifications (PSAR)

An application for a construction permit should include preliminary Technical Specifications that identify and provide justification for the selection of variables, conditions, or other limitations that are determined to be probable subjects of the final Technical Specifications. Special attention should be given to those areas that influence the final design in order to minimize later facility modifications to accommodate conditions of the final Technical Specifications. In particular, this review should determine the design suitability of those features and specifications that affect the type, capacity, and number of safety-related systems and the capability for performance of surveillance activities involving those safety-related systems.

The preliminary Technical Specifications and bases should be included in this chapter of the PSAR. The submittal should be consistent with the format and content of the NRC Standard Technical Specification for the appropriate NSSS vendor. Each specification should be as complete as possible and should include preliminary numerical values, graphs, tables, and other data. References to the applicable sections of the PSAR that support the bases and provide clarifying details for each specification should be supplied. Justification should be provided for deletions from or additions to the Standard Technical Specifications pertinent to the selected NSSS vendor.

16.2 Proposed Final Technical Specifications (FSAR)

The Technical Specifications submitted in support of an operating license application should be the finalized version of those specifications originally included in the PSAR. The numerical values, graphs, tables, and other data should reflect the final refinements in design, results of tests or experiments, and expected method of operation. This information should be included in this chapter of the FSAR.

17. QUALITY ASSURANCE

In order to provide assurance that the design, construction, and operation of the proposed nuclear power plant are in conformance with applicable regulatory requirements and with the design bases specified in the license application, it is necessary that a quality assurance (QA) program be established by the applicant. In this chapter of the SAR, the applicant should provide a description of the QA program to be established and executed during the design, construction, preoperational testing, and operation of the nuclear power plant. The QA program must be established at the earliest practical time consistent with the schedule for accomplishing the activity. Where some portions of the QA program have not yet been established at the time the SAR is prepared because the activity will be performed in the future, the description should also provide a schedule for implementation. The program must meet the requirements of Appendix B to 10 CFR Part 50. The inspection and survey systems required by §50.55a, "Codes and Standards," of 10 CFR Part 50 may be used in partial fulfillment of these requirements to the extent that they are shown by the description of the QA program to satisfy the applicable requirements of Appendix B.

In order to facilitate the presentation of the information, the QA program for each of the major organizations involved in executing the QA program should include the information described (either separately for each organization or integrally for all organizations) in accordance with the outline presented below. It is not intended to dictate the format of any QA Program Manual; that is left to the discretion of the applicant. It is required, however, that the description in the SAR address, at a minimum, each of the criteria in Appendix B in sufficient detail to enable the reviewer to determine whether and how all the requirements of the appendix will be satisfied in accordance with §50.34 of 10 CFR Part 50. Reference to appropriate portions of other sections of the SAR may suffice.

NRC regulatory guides and the documents entitled "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," (WASH 1283), "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," (WASH 1309), and "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants," (WASH 1284) contain guidance on acceptable methods of implementing portions of the quality assurance program.* The SAR should specifically indicate whether this guidance will be followed. If such guidance will not be followed, the SAR should describe specific alternative methods that will be used and the manner of implementing them and should identify the organizations responsible for their implementation.

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*WASH 1283, 1284, and 1309 contain a number of draft standards. As these draft standards are issued as approved American National Standards, it is expected that they will be endorsed by regulatory guides. The applicability of the regulatory guide versus the draft standard will be addressed in the implementation section of each guide or in amendments to this Standard Format

143 284

Where a portion of the QA program to be implemented will follow the guidance provided by a regulatory guide, WASH 1283, WASH 1309, or WASH 1284, the program description may consist of a statement that the guidance will be followed for that portion of the QA program. When these documents are used in describing the QA program, the applicant should indicate how the guidance documents will be applied to portions of the QA program and should delineate the organizational element responsible for implementing various provisions of the respective guidance documents within each major organization in the project, including that of the applicant, the architect-engineer, the nuclear steam system supplier, the constructor, the construction manager (if other than the constructor).

17.1 Quality Assurance During Design and Construction

17.1.1 Organization

17.1.1.1. The PSAR should describe clearly the authority and duties of persons and organizations performing the QA functions of assuring that the QA program is established and executed and of verifying that an activity has been correctly performed. The PSAR should provide organization charts and functional responsibility descriptions that denote the lines of responsibility and areas of authority within each of the major organizations in the project, including those of the applicant, the architect-engineer, the nuclear steam system supplier, the constructor, and the construction manager (if other than the constructor). These charts and descriptions should present the structure of QA organizations involved as well as other functional organizations performing activities affecting quality in design, procurement, manufacturing, construction and installation, testing, inspection, and auditing with clear delineation of their responsibility, authority, and relationship to corporate management. In addition, a single overall project organization chart should be included showing how the major organizations or companies working directly for the applicant on the project interrelate with one another.

17.1.1.2. The PSAR should describe the level of management responsible for establishing the QA policies, goals, and objectives and should describe the continuing involvement of this management level in QA matters. The PSAR should tell what position has overall authority and responsibility for the QA program and tell what position is responsible for final review and approval of the QA program and related manuals. The qualification requirements of the principal QA and quality control positions should be described.

17.1.1.3. The PSAR should describe those measures which assure that persons and organizations performing QA functions have sufficient authority and organizational freedom to (1) identify quality problems, (2) initiate, recommend, or provide solutions, and (3) verify implementation of solutions. The PSAR should describe the measures which assure that persons and organizations assigned the responsibility for checking, auditing, inspecting, or otherwise verifying that an activity has been correctly performed report to a management level such that this required authority

143 285

and organizational freedom, including sufficient independence from the pressures of production, are provided. Irrespective of the organizational structure, the PSAR should describe how the individual or individuals with primary responsibility for assuring effective implementation of the QA program at any location where activities subject to the control of the QA program are being performed will have direct access to such levels of management as may be necessary to carry out this responsibility. The PSAR should indicate from whom the persons performing QA functions receive technical direction for performing QA tasks and administrative control (salary review, hire/fire, position assignment). The PSAR should identify those positions or organizations which have written delegated responsibility and authority to stop work or control further processing, delivery, installation, or use of nonconforming items until proper disposition of the deficiency has been approved.

The PSAR should describe how requirements will be imposed on contractors and subcontractors to assure that individuals or groups within their organizations performing QA functions have sufficient authority and organizational freedom to effectively implement their respective QA programs.

17.1.1.4. The PSAR should describe the extent to which the applicant will delegate to other contractors the work of establishing and executing the QA program or any part thereof. A clear delineation of those QA functions which are implemented within the applicant's QA organization and those which are delegated to other organizations should be provided in the PSAR. The PSAR should describe the method by which the applicant will retain responsibility for and maintain control over those portions of the QA program delegated to other organizations and should identify the organization responsible for verifying that delegated QA functions are properly carried out. The PSAR should identify major work interfaces for activities affecting quality and describe how clear and effective lines of communication exist between the applicant and his principal contractors to assure necessary coordination and control of the QA program.

17.1.2 Quality Assurance Program

17.1.2.1. The QA program in the PSAR should cover each of the criteria in Appendix B to 10 CFR Part 50 in sufficient detail to permit a determination as to whether and how all of the requirements of Appendix B will be satisfied. The PSAR should (1) describe the extent to which the QA program will conform to various provisions of WASH 1283, WASH 1309, and regulatory guides that provide guidance on acceptable methods of implementing portions of the QA program and (2) identify the organizational element responsible for implementing these provisions. If the applicant elects not to follow the above guidance, the PSAR should describe in detail equivalent to that furnished in the NRC guidance the alternative methods that will be used and the manner of implementing them and should indicate the organizations responsible for their implementation.

17.1.2.2. The PSAR should identify the safety-related structures, systems, and components to be controlled by the QA program.

17.1.2.3. The PSAR should describe the measures which assure that the QA program is being established at the earliest practicable time consistent with the schedule for accomplishing activities affecting quality for the project. That is, the PSAR should describe how the QA program is being established in advance of the activity to be controlled and how it will be implemented as the activity proceeds. Those activities affecting quality initiated prior to the submittal of the PSAR, such as establishing information required to be included in the PSAR, design and procurement, safety-related site testing and evaluation, and preparation activities should be identified in the PSAR. The PSAR should describe how these activities are controlled by a QA program which complies with Appendix B to 10 CFR Part 50.

17.1.2.4. The PSAR should describe how the QA program is documented by written policies, procedures, or instructions and how it will be implemented in accordance with these policies, procedures, or instructions. The PSAR should include a listing of QA program procedures or instructions that will be used to implement the QA program for each major activity such as design, procurement, construction, etc. The procedure list should identify which criteria of Appendix B to 10 CFR Part 50 are implemented by each procedure. In the event certain required procedures are not yet established, a schedule for their preparation should be provided in the PSAR.

17.1.2.5. The PSAR should summarize the corporate QA policies, goals, and objectives; and it should describe how disputes involving quality are resolved.

17.1.2.6. The PSAR should describe the program that provides adequate indoctrination and training of personnel performing activities affecting quality to assure that suitable proficiency is achieved and maintained. The PSAR should describe how the indoctrination and training program will assure that:

1. Personnel performing activities affecting quality are appropriately trained in the principles and techniques of the activity being performed,

2. Personnel performing activities affecting quality are instructed as to purpose, scope, and implementation of governing manuals, policies, and procedures,

3. Appropriate training procedures are established, and

4. Proficiency of personnel performing activities affecting quality is maintained.

143 287

17.1.2.7. The PSAR should describe the qualification requirements for the position or positions responsible for assuring effective implementation of the QA program of the applicant and of his major contractors.

17.1.2.8. The PSAR should describe the measures that assure that activities affecting quality will be accomplished under suitable controlled conditions, including (1) the use of appropriate equipment, (2) a suitable environment for accomplishing the activity, e.g., adequate cleanliness, and (3) compliance with necessary prerequisites for the given activity.

17.1.2.9. The PSAR should describe the measures that assure that there is regular management review of the QA program to assess its effectiveness and the adequacy of its scope, and implementation. The PSAR should describe the provisions for reviews by management above or outside the QA organization to assure achieving an objective program assessment.

The PSAR should describe the measures that assure that the QA organization of the applicant will (1) review and document agreement with the QA programs of the principal contractors and (2) conduct or have conducted audits of the contractors' QA program activities.

17.1.2.10. The PSAR should provide a summary description of advanced planning that demonstrates control of quality-related activities including management and technical interfaces between the contructor, the architect-engineer, the nuclear steam system supplier, and the applicant during the phaseout of design and construction and during preoperational testing and plant turnover.

17.1.2.11. The PSAR should describe provisions for maintaining the QA program description current.

17.1.3 Design Control

17.1.3.1. The PSAR should describe the design control measures that assure that (1) applicable regulatory requirements and design bases for safety-related structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions, (2) appropriate quality standards are specified in design documents, and (3) deviations from such standards are controlled.

17.1.3.2. The PSAR should describe measures that assure that adequate review and selection for application suitability is conducted for materials, parts, equipment, and processes that are essential to safety-related functions of the structures, systems, and components. The PSAR should describe provisions that assure that standard commercial or so-called "off the shelf" materials, parts, and equipment also receive adequate application review and selection.

17.1.3.3. The PSAR should describe the program for applying design control measures to such aspects of design as reactor physics; stress, thermal, hydraulic, and accident analysis; materials compatibility; and accessibility for maintenance, inservice inspection, and repair and should describe measures for delineation of acceptance criteria for inspections and tests.

17.1.3.4. The PSAR should describe measures that assure verification or checking of design adequacy, such as design reviews, use of alternative calculational methods, or performance of a qualification testing program under the most adverse design conditions. The PSAR should identify the positions or organizations responsible for design verification or checking and should describe measures that assure that the verifying or checking process is performed by individuals or groups other than those who performed the original design, but who may be from the same organization.

17.1.3.5. The PSAR should describe measures for identifying and controlling design interfaces, both internal and external, and for coordination between participating design organizations. The PSAR should describe measures in effect between participating design organizations for review, approval, release, distribution, collection, and storage of documents involving design interfaces and changes thereto. The PSAR should describe how these measures will assure that these design documents are controlled in a timely manner to prevent inadvertent use of superseded design information.

17.1.3.6. The PSAR should describe the measures that will be employed to assure that design changes, including field changes, are subject to the same design controls that were applied to the original design and are reviewed and approved by the organization that performed the original design unless the originating organization designates another responsible organization.

17.1.4 Procurement Document Control

17.1.4.1. The PSAR should describe measures that assure that documents, and changes thereto, for procurement of material, equipment, and services, whether purchased by the applicant or the contractors or subcontractors, correctly include or reference the following as necessary to achieve required quality:

1. Applicable regulatory, code, and design requirements,

2. Quality assurance program requirements,

3. Requirements for supplier documents such as instructions, procedures, drawings, specifications, inspection and test records, and supplier QA records to be prepared, submitted, or made available for purchaser review or approval,

4. Requirements for the retention, control, and maintenance of supplier QA records,

143 289

5. Provision for purchaser's right of access to suppliers' facilities and work documents for inspection and audit, and

6. Provision for supplier reporting and disposition of nonconformances from procurement requirements.

17.1.4.2. The PSAR should describe (1) measures that clearly delineate the control responsibilities and action sequence to be taken in the preparation, review, approval, and issuance by competent personnel of procurement documents and (2) measures that assure that changes or revisions of procurement documents are subject to the same review and approval requirements as the original documents.

17.1.4.3. The PSAR should describe measures that assure (1) that procurement documents require suppliers to have and implement a documented QA program for purchased materials, equipment, and services to an extent consistent with their importance to safety, (2) that the purchaser has evaluated the supplier before the award of the procurement order or contract to assure that the supplier can meet the procurement requirements, and (3) that procurement documents for spare or replacement items will be subject to controls at least equivalent to those used for the original equipment.

17.1.5 Instructions, Procedures, and Drawings

17.1.5.1. The PSAR should describe measures that assure that activities affecting quality such as design, procurement, manufacturing, construction and installation, testing, inspection, and auditing are prescribed by appropriately documented instructions, procedures, or drawings and that these activities will be conducted in accordance with these documents.

17.1.5.2. The PSAR should describe the system whereby the documented instructions, procedures, and drawings will include appropriate quantitative (such as dimensions, tolerances, and operating limits) and qualitative (such as workmanship samples and weld radiographic acceptance standards) acceptance criteria for determining that prescribed activities have been satisfactorily accomplished.

17.1.6 Document Control

17.1.6.1. The PSAR should describe those measures established to control the issuance of documents such as instructions, procedures, and drawings, including changes thereto, that prescribe all activities affecting quality. The description should cover control measures that assure that:

1. Documents are reviewed for adequacy (i.e., information is clearly and accurately stated) and are approved by authorized personnel for issuance and use at locations where the prescribed activity will be performed before the activity is started,

143 290

2. Means such as use of updated master document lists exist to assure that obsolete or superseded documents are replaced in a timely manner by updated applicable document revisions, and

3. Document changes are reviewed and approved by the same organizations that performed the original review and approval unless delegated by the originating organization to another responsible organization.

17.1.6.2. The PSAR should identify the types of documents to be controlled and the group responsible for review, approval, and issuance of documents and changes thereto.

17.1.7 Control of Purchased Material, Equipment, and Services

17.1.7.1. The PSAR should describe those measures that assure that material, equipment, and services purchased directly by the applicant or his contractors and subcontractors will conform to procurement document requirements. The PSAR should describe the measures that provide, as appropriate, for:

1. Evaluation and selection of sources of supply before the award of the procurement order or contract,

2. Surveillance at the supplier's facility by the purchaser or his representative in accordance with written procedures during design, manufacture, inspection, and test of the procured item or service to verify compliance with quality requirements,

3. Source and/or receipt inspection in accordance with written procedures and acceptance criteria of procured items furnished by the supplier,

4. Documentary evidence at the site from the supplier that procured items meet procurement quality requirements such as codes, standards, or specifications. The PSAR should describe measures established by the applicant to (a) examine and indicate acceptance of this documented evidence during source or receipt inspection and (b) assure that this documented evidence is available at the nuclear power plant site prior to installation or use of the procured item and that the documentation will be retained at the plant site, and

5. Periodic verification of supplier's certificates of conformance to assure that they are meaningful.

17.1.7.2. The PSAR should describe measures whereby the applicant or his designated representative will audit and evaluate the effectiveness of the control of quality-related activities of contractors and subcontractors at a frequency and extent consistent with the importance to safety, complexity, and quantity of the item or service being furnished.

17.1.8 Identification and Control of Materials, Parts, and Components

The PSAR should describe measures established to identify and control items such as materials, parts, and components, including partially fabricated assemblies, to prevent use of incorrect or defective items. The PSAR should describe measures that assure (1) that identification of the item, (i.e., heat number, part number, serial number, or other appropriate marking) is maintained either on the item or on records traceable to the item and verified, as required, throughout fabrication, erection, installation, and use of the item and (2) that the method and location of the identification does not affect the function or quality of the item being identified.

17.1.9 Control of Special Processes

The PSAR should describe measures established to control special processes such as welding, heat treating, nondestructive testing, and electrochemical machining and to assure that they are accomplished by qualified personnel using written procedures qualified in accordance with applicable codes, standards, specifications, or other special requirements. The PSAR should describe those measures that assure that qualifications of special processes, personnel performing special processes, and equipment are kept current and that record files thereof are maintained.

17.1.10 Inspection

17.1.10.1. The PSAR should describe the measures that assure that a program for inspection is established and implemented by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The PSAR should describe measures that assure that (1) inspection personnel are appropriately qualified and are independent of the individual or group performing the activity being inspected, (2) inspections or tests are performed for each work operation as necessary to verify quality, (3) indirect control by monitoring processing methods, equipment, and personnel is used if direct inspection of processed material or products is impossible or disadvantageous, and (4) both inspection and process monitoring are used when control is inadequate without both. The PSAR should describe measures that assure that (1) inspection procedures and instructions are made available with necessary drawings and specifications for use prior to performing the inspections, (2) inspectors' qualifications or certifications are kept current, (3) replaced or reworked items are inspected in accordance with original inspection requirements, and (4) modified or repaired items are inspected by methods that are equivalent to the original inspection method.

17.1.10.2. The PSAR should describe the system whereby appropriate documents will identify any mandatory inspection holdpoints that require witnessing or inspecting by the applicant or his designated representative and beyond which work may not proceed without the consent of his designated representative.

143 292

17.1.11 Test Control

17.1.11.1. The PSAR should describe the measures that establish a test program that (1) identifies all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service, (2) is conducted by trained and appropriately qualified personnel in accordance with written test procedures that incorporate or reference the requirements and acceptance limits contained in applicable design documents, and (3) includes testing that will be performed under the construction permit.

17.1.11.2. The PSAR should describe the measures that assure test procedures have provisions for assuring that:

1. All prerequisites for the given test have been met,

2. Adequate test instrumentation and equipment are available, and

3. The test is performed under suitable environmental conditions and with adequate test methods.

17.1.11.3. The PSAR should describe the system whereby test results are documented and evaluated to assure that test requirements have been satisfied.

17.1.12 Control of Measuring and Test Equipment

The PSAR should describe the measures established to assure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly identified, controlled, adjusted, and calibrated at specified periods to maintain accuracy within necessary limits. The PSAR should describe measures that assure (1) that these devices are adjusted and calibrated against certified equipment or reference or transfer standards having known valid relationships to nationally recognized standards or (2) that if no national standards exist, the basis for calibration is documented. The PSAR should describe the measures that assure that the error of calibration standards is less than the error of production measuring and test equipment. The PSAR should describe provisions that will apply if measuring and test equipment is found out of calibration (1) for evaluating the validity of previous inspection or test results and the acceptability of items inspected or tested since the last calibration check and (2) for repeating original inspections or tests using calibrated equipment where necessary to establish acceptability of suspect items. The PSAR should describe measures that assure the maintenance of records that indicate the calibration status of all items under the calibration system and that identify the measuring and test equipment.

143 293

17.1.13 Handling, Storage, and Shipping

The PSAR should describe the measures established to control the handling, storage, shipping, cleaning, and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. The PSAR should describe the measures for specifying and providing, when necessary for particular products, special protective environments such as inert gas atmosphere, specific moisture content levels, and temperature levels.

17.1.14 Inspection, Test, and Operating Status

The PSAR should describe measures established to indicate by the use of markings such as stamps, tags, labels, routing cards, or other suitable means the status of inspections and tests performed on individual items of the nuclear power plant throughout fabrication, installation, and test. The PSAR should describe measures that provide for the identification of items that have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of such inspections and tests. The PSAR should describe the measures established for indicating the operating status of structures, systems, and components of the nuclear power plant such as tagging valves and switches to prevent inadvertent operation.

17.1.15 Nonconforming Materials, Parts, or Components

The PSAR should describe the measures established to control materials, parts, or components that do not conform to requirements in order to prevent their inadvertent use or installation. The PSAR should describe measures that provide for, as appropriate, identification, documentation, segregation, disposition, and notification to affected organizations. The PSAR should describe measures that assure that nonconforming items are reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures. The PSAR should describe measures that control further processing, delivery, or installation pending proper disposition of the deficiency. The PSAR should describe measures established by the applicant (1) for contractors to report to him those nonconformances concerning departures from procurement requirements that are dispositioned "use as is" or "repair" and (2) to make such nonconformance reports part of the documentation required at the nuclear plant site or to include a description of the nonconformance and its disposition on certificates of conformance that are provided to the site prior to installation or use of material or equipment at the site. The PSAR should state whether periodic analyses of nonconformance reports are performed to show quality trends and whether such analyses are forwarded to management.

143 294

17.1.16 Corrective Action

17.1.16.1. The PSAR should describe the measures that assure that conditions adverse to quality such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

17.1.16.2. The PSAR should describe how, in the case of significant conditions adverse to quality, the cause of the condition is determined, corrective action is taken to preclude repetition, and the problem with its determined cause and corrective action is documented and reported to appropriate levels of management.

17.1.17 Quality Assurance Records

17.1.17.1. The PSAR should describe the measures that assure that sufficient records are maintained to furnish evidence of activities affecting quality. The PSAR should describe how the content of such records (1) includes at least the following: test logs; results of reviews, drawings, inspections, tests, audits, monitoring of work performance, and materials analyses; and such data as qualifications of personnel, procedures, and equipment; (2) identifies the type of operation, the inspector or data recorder, the results, the acceptability, and action taken in connection with any deficiencies noted; and (3) provides sufficient information to permit identification of the record with the item or activity to which it applies.

17.1.17.2. The PSAR should describe the measures that assure that records will be identifiable and retrievable.

17.1.17.3. The PSAR should describe the measures that establish requirements (consistent with regulatory requirements and responsibilities concerning record submittal and retention, security, and storage facilities) for protecting records from destruction by fire, flooding, tornadoes, insects, and rodents and from deterioration by extremes in temperature and humidity.

17.1.18 Audits

The PSAR should describe the program of the applicant and of the principal contractors for conducting comprehensive planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program.

The PSAR should describe the program features that cover the functions listed below and should identify the positions or organizations that perform these functions.

1. External audits to be performed by the applicant and his principal contractors on their respective suppliers,

2. Internal audits to be performed by the applicant and his principal contractors within their respective organizations,

3. The planning and scheduling of audits to assure that they are regularly scheduled on the basis of the status and safety importance of the activities being performed and are initiated early enough to assure effective quality assurance during design, procurement, manufacturing, construction and installation, inspection, and testing,

4. Conduct of audits in accordance with written procedures or checklists by appropriately trained and qualified personnel not having direct responsibility in the area being audited, and

5. Documentation of audit results with review by management responsible for the area audited and, where indicated, followup action taken, including re-audit of the deficient areas.

17.2 Quality Assurance (QA) During the Operations Phase

The FSAR should describe the QA program that will assure the quality of all safety-related items and activities during the operations phase. These activities include plant operation, maintenance, repair, inservice inspection, refueling, modifications, testing, and inspection under the operating license.

The description of the QA program in the FSAR should include the applicable information requirements outlined in Section 17.1 (i.e., substitute "FSAR" for "PSAR" in 17.1, above), except for those activities applicable only to the construction phase (activities performed under the construction permit). The FSAR should describe the QA program under the cognizance of the offsite and onsite QA organizations and should show that it addresses each of the criteria of Appendix B to 10 CFR Part 50. The description should delineate any significant differences in functional responsibilities between the offsite and onsite QA organizations.

The FSAR should describe the extent to which the operations phase QA program will follow the guidance in WASH-1284, "Guidance on QA Requirements During the Operations Phase of Nuclear Power Plants," and the extent to which activities involving design, procurement, and construction during the operations phase will follow the guidance in WASH-1283, "Guidance on QA Requirements During Design and Procurement Phase of Nuclear Power Plant," and in WASH-1309, "Guidance on QA Requirements During the Construction Phase of Nuclear Power Plants." If such guidance will not be followed, the applicant should describe acceptable alternative methods in detail equivalent to that furnished in the above guidance.

143 296

APPENDIX A*

INTERFACES FOR STANDARD DESIGNS

Table of Contents

FIGURES

TABLES

*Although Appendix A is a new addition to Regulatory Guide 1.70, Revision 3, the lines indicating changes have not been included.

ABBREVIATIONS FOR NUCLEAR PLANT SYSTEMS

ADS	-	Automatic depressurization system
BOP	-	Balance of plant (all systems, structures, and components comprising a total plant excluding the NSSS and site- and utility-specific items)*
CCWS	-	Component cooling water system
CIS	-	Containment isolation system
CVCS	-	Chemical and volume control system
ECCS	-	Emergency core cooling systems
ESF	-	Engineered safety feature
GWMS	-	Gaseous waste management system
I&C	-	Instrumentation and control
LWMS	-	Liquid waste management system
MSIVLCS	-	Main steam isolation valve leakage control system
MSLIV	-	Main steam line isolation valve
NSSS	-	Nuclear steam supply system (components and piping comprising the RCS and directly related auxiliary systems)*
PRDS	-	Pressurizer relief discharge system
RCICS	-	Reactor core isolation cooling system
RCPB	-	Reactor coolant pressure boundary
RCS	-	Reactor coolant system
RHRS	-	Residual heat removal system
RWCUS	-	Reactor water cleanup system
SSWS	-	Station service water system
SWMS	-	Solid waste management system

*See Amendment 1 to WASH 1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants."

143 298

INTERFACES FOR STANDARD DESIGNS

I. INTRODUCTION

Safety-related interfaces must be identified and defined for standard designs submitted under Option 1 (Reference Systems) of the Commission's standardization policy to establish the requirements that must be met and assumptions that must be verified by other unspecified portions of a nuclear plant design to ensure that systems, components, and structures within the standard design will perform their safety functions. Safety-related interfaces also include information that may be useful in the design and staff review of the unspecified portions of the plant design. The safety functions of a standard design are those essential functions that ensure (1) the integrity of the reactor coolant pressure boundary; (2) that the specified acceptable fuel design limits are not exceeded as a result of anticipated transients; (3) the capability to shut down the reactor and maintain it in a safe shutdown condition; and (4) the capability to prevent or mitigate the consequences of an accident that could result in radiation exposures in excess of applicable guidelines. Interfaces are used, therefore, to provide a basis for ensuring that the matching portions of a nuclear plant design, as described in a PSAR for a CP application that references the standard design or in another Standard Safety Analysis Report (SSAR) for a matching portion of the plant, are compatible with the standard design regarding the safety-related aspects of the plant design.

This appendix describes safety-related interfaces, for light-water reactors only, that should be presented at the preliminary design* stage of review by the reactor vendor in a Nuclear Steam Supply System SSAR (NSSS-SSAR)** and by the architect-engineer in a Balance-of-Plant SSAR (BOP-SSAR).** The interfaces for a BOP-SSAR are also directly applicable to an SSAR describing an entire nuclear plant (NSSS plus BOP, but excluding utility- and site-specific items). This appendix also describes an acceptable format for presenting interfaces in an SSAR.

Criteria for determining the acceptability of interfaces, as necessary for safety, are not included in this appendix. While not identified specifically as interface acceptance criteria, the criteria are part of other guidance already made available by the NRC, including that contained in the regulations, regulatory guides, and codes and standards.

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* Many of the interfaces identified in this document are also applicable at the final design stage of review. Definitive guidance regarding all final design interfaces will be provided later as the need arises.

** The specific interface items presented herein apply only to an NSSS/BOP division of design scope for a nuclear plant or to an entire nuclear plant (NSSS and BOP); they do not apply to any other division of design scope such as a nuclear island/turbine island.

143 299

The compilation of interfaces presented in this appendix is based on staff consideration of all safety criteria applicable to the review of nuclear power plant designs, including those contained within the regulations, regulatory guides, and codes and standards, and on the background and experience acquired during the staff review of the several standard design applications already submitted. In this light, the staff considers the present listing of interfaces to be essentially complete and to promote maximum flexibility for design and for component selection consistent with the requirements for safety. However, standard design applications and utility applications referencing standard designs should not necessarily be limited to the interfaces listed; any additional interfaces determined to be important to safety should be identified and addressed in these applications, especially those interfaces that may be unique to a particular plant design. It is also the staff's intent to supplement and revise the interface lists, as well as other aspects of this appendix, as additions and modifications are indicated.

II. SOURCES OF INTERFACES

Interfaces for standard designs stem from the following sources:

a. Requirements for safe operation of the standard design that must be satisfied by matching portions of the plant design or by the utility (e.g., cooling water and electric power requirements for the NSSS that must be provided by the BOP, an inservice inspection program for the NSSS and BOP that must be provided by the utility).

b. Assumptions made for the standard design that must be more precisely defined during the design coordination effort between the reactor vendor and the architect-engineer or between the architect-engineer and the utility (e.g., mass and energy release rates during a LOCA specified by the reactor vendor that must be coordinated with the containment design provided by the architect-engineer).

c. Site-related design assumptions upon which the standard design is based.

d. Criteria pertinent to the standard design described in the SSAR under review that may be useful for the design and staff review of matching systems, components, and structures (i.e., safety criteria for the items within the standard design, including codes and standards, General Design Criteria, and regulatory guides).

Each of the above sources was used by the staff in preparing the lists of interfaces shown in Sections VI and VII and, in turn, should be used by reactor vendors and architect-engineers when identifying and defining interfaces for presentation in SSARs.

143 300

III. INTERFACES TO BE ADDRESSED IN SSARs

The interface items that should be addressed in SSARs for both normal and abnormal operating conditions have been identified by the staff as shown in Sections VI and VII. Those interfaces listed in Section VI should be defined by reactor vendors in NSSS-SSARs. Those listed in Section VII should be defined by architect-engineers in BOP-SSARs or by other organizations in standard plant SSARs (NSSS plus BOP). In addressing these interfaces, the standard design applicant should clearly define the scope of design encompassed by the SSAR; the definition so provided should be consistent with the gross definition of the content of NSSS-SSARs and BOP-SSARs as given in Amendment 1 to WASH-1341. The sources of the interfaces listed in Sections VI and VII are items a, b, c, and d described in Section II above.

IV. USE OF INTERFACES PRESENTED IN SSARs

All interfaces presented in an SSAR should be addressed in a referencing SAR (either a PSAR for a CP application or another SSAR) that describes the matching portion of the nuclear plant design. The description of the matching portion should clearly indicate that each interface has been recognized, used, and satisfied by the design of the interfacing system, component, or structure. For those interfaces involving a design coordination effort between the reactor vendor and the architect-engineer, the utility application should clearly describe the outcome of this effort in terms of the resulting design of the interfacing systems, components, and structures. In this way, the compatibility of matching portions of a plant design with regard to licensing requirements is demonstrated.

For site interfaces, the utility PSAR referencing a standard design should demonstrate that the site design parameters established as the basis for the standard design envelop the characteristics of the proposed plant site described in the CP application (e.g., the response spectrum used for the seismic design of the standard design applied at the foundation level should be shown to envelop the response spectrum derived at the foundation level for the proposed site).

It should be noted that acceptability to the staff of the compatibility of matching portions of a plant design with regard to licensing requirements in no way relieves the utility-applicant that has referenced a standard design in his application for licenses from his responsibility under the NRC regulations to ensure that all interfaces between matching systems, components, and structures are satisfied for compatibility.

143 301

V. FORMAT FOR INTERFACE PRESENTATION IN SSARs

Interfaces should be presented in an SSAR in a manner that will facilitate their location by staff reviewers and other groups involved in the licensing and design processes. In addition, interfaces should be presented on a system-by-system basis consistent with the approach for presenting plant design information established in Regulatory Guide 1.70. The following guidance for presenting interfaces in SSARs describes an acceptable format to accomplish these purposes:

a. Chapter 1, "Introduction and General Description of Plant," should include an interface section presenting an overall "road map" matrix to guide the reviewer to other sections in the SSAR where the specific interfaces can be found. The matrix should include:

1. A listing of all systems and structures within the standard design that interface with matching unspecified portions of the plant;

2. A listing of other interface areas that can be referenced in support of the items listed in 1 above;

3. A listing of the particular items in the matching unspecified portion with which the standard design items interface (e.g., the RHRS in the NSSS-SSAR interfaces with the CCWS, emergency onsite power system, containment sump, and refueling water storage tank in a BOP-SSAR or a PSAR); and

4. Identification of the section in the SSAR in which the specific interfaces are described.

Examples of an acceptable approach for preparing the matrices for an NSSS-SSAR and a BOP-SSAR are shown in Figures 1 and 2, respectively.

b. Specific interfaces should be presented on a system-by-system basis to the maximum extent practicable and should be shown in a separate subsection (as identified in Sections VI and VII and in the Table of Contents for Regulatory Guide 1.70) directly associated with the system description (not in the section assigned by Regulatory Guide 1.70 for the system description). The subsection should incorporate drawings, piping and instrumentation diagrams, and tables either directly or by reference (provided the interfaces intended to be referenced are clearly indicated therein). In general, descriptive material in other sections that may contain interfaces should not be referenced.

c. Interfaces of a broader nature that apply to classes of systems, components, or structures (e.g., items 3.4.3, 3.5.4, 3.6.3, 3.7.5, 3.8.6, 3.9.7, 3.10.5, and 3.11.6 in Section VI for an NSSS-SSAR and items 3.3.3, 3.4.3, 3.5.4, 3.7.5, 3.8.6, and 3.9.7 in Section VII for a BOP-SSAR) should be presented in the appropriate sections of other chapters of the SSAR (i.e., Chapter 3). These are supporting interfaces that should, in turn, be referenced in the interface subsections for the systems and structures

Figure 1

EXAMPLE OF MATRIX OF INTERFACE AREAS FOR AN NSSS-SSAR

Items on Matching Portion of Plant

NSSS Interface Areas	Feedwater System	Main Steam System	Component Cooling Water System	Offsite Power System	Onsite A.C. Power System	Auxiliary F.W. System	Containment	Site	Ventilation System	Liquid Waste Management System	Control Room	D.C. Power Supply	Location in SSAR
1. SYSTEM INTERFACE AREAS													
Reactor Coolant Pressure Boundary	X	X	X	X		X	X				X		5.2.6
Emergency Core Cooling Systems			X	X	X		X				X	X	6.3.6
Reactor Trip System		X		X	X		X				X	X	7.2.3
Habitability Systems			X								X		6.4.7
Fuel Handling System							X		X				9.1.4.6
Standby Liquid Control System				X	X						X	X	9.3.5.6
Chemical and Volume Control System			X	X					X	X	X		9.3.4.6
Reactor Water Cleanup System			X	X					X	X	X		5.4.8.4
Residual Heat Removal System			X	X	X		X		X		X	X	5.4.7.5
2. SUPPORTING INTERFACE AREAS													
Flood Protection													3.4.3
Missile Protection													3.5.4
Pipe Whip Protection													3.6.3
Mechanical Systems and Components													3.9.7
Environmental Design of Mechanical and Electrical Equipment													3.11.6
Inservice Inspection of Class 2 & 3 Components													6.6.9
Fire Protection													9.5.1.6
Safety Actions by BOP													15.X.X

143 303

Figure 2

EXAMPLE OF MATRIX OF INTERFACE AREAS FOR A BOP-SSAR

Site- and Utility-Specific Items

BOP Interface Areas	Switchyard	Ultimate Heat Sink	Intake Structure	Inservice Insp. Program	Initial Test Program	Industrial Security	10 CFR Part 50 App. I	Meteorology	Seismic Design Parameter	Wind & Tornado Parameter	Geology	Probable Maximum Flood	Location in SSAR
1. INTERFACE AREAS FOR SYSTEMS, COMPONENTS, AND STRUCTURES													
Station Service Water System		X	X	X	X	X			X			X	9.2.1
Instrumentation and Control			X		X				X				7.8
Fire Protection Program	X	X	X		X				X				9.5.1.6
Onsite A.C. Power System	X		X		X	X			X				8.3.1.5
Water Systems		X	X	X	X								9.2.7
Liquid Waste Management System		X		X	X		X						11.2.4
Gaseous Waste Management System				X	X		X	X					11.3.4
Effluent Monitoring and Sampling								X					11.5.3.X
Other Auxiliary Systems		X		X	X								9.5.9
2. SUPPORTING INTERFACE AREAS													
Wind and Tornado Loadings										X			3.3.3
Water Level Design												X	3.4.3
Seismic Design									X				3.7.5
Design of Category I Structures								X	X	X	X	X	3.8.6
Industrial Security						X							13.6.3
Mechanical Systems and Components				X	X								3.9.7

143 304

described in the SSAR. Examples include site design parameters, protection against missiles, and protection against pipe whip.

d. For an NSSS-SSAR, the interfaces identified in Section VI should be presented for each system (review area) in the following categories to facilitate review. Review areas in Section VI of a broad nature, as discussed in item c above, need not use these categories for presentation of interfaces.

1. Power - Requirements for all types of power for safety systems and components.

2. Protection Against Natural Phenomena - Requirements for protection of safety-related systems and components against naturally occurring events such as earthquakes, wind, tornadoes, and floods.

3. Protection Against Effects of Pipe Failure - Requirements for protection of safety-related systems, components, and structures inside and outside containment against the dynamic effects resulting from the failure of piping in high- and moderate-energy systems, including pipe whip, jet impingement, and other dynamic effects.

4. Missiles - Requirements for protection of reactor coolant pressure boundary and other safety-related systems against internally generated missiles and missiles generated by naturally occurring events both inside and outside containment; identification of potential missiles from NSSS equipment.

5. Separation - Requirements for physical separation to prevent a single event from causing failure of redundant safety systems and components.

6. Independence - Requirements for independence to prevent a failure in a safety system or component from causing failure in its redundant safety system or component.

7. Thermal Limitations - Requirements for heating or cooling of safety systems and components, including fluid conditions and limitations.

8. Monitoring - Requirements for performance surveillance, testing, and inspection of safety systems and components (technical specifications not included).

9. Actuation/Controls - Requirements for actuation of safety systems and components, for control of their subsequent operation, and for interlocks.

10. Chemistry/Sampling - Requirements for fluid chemistry, purity, and sampling for safety systems and components.

143 305

11. Materials - Requirements for materials for safety systems and components.

12. System/Component Arrangement - Location requirements (including inservice inspection and testing) that safety systems and components place on plant arrangement.

13. Radioactive Waste - Source term characteristics for the collection, treatment, and disposal of radioactive wastes.

14. Related Service - Requirements for other essential services for safety systems and components (e.g., interfaces from Section V.f below, fire protection, compressed air).

15. Overpressure Protection - Requirements for ensuring that pressure limits for safety systems are not exceeded.

16. Environment - Requirements for environmental conditions that must be provided for proper operation of safety systems and components.

17. Mechanical Interaction Between Systems - Requirements for consideration of differential motion, including seismic effects and thermal expansion.

18. Design Criteria - Criteria upon which the NSSS system designs, or portions thereof, are based.

e. For a BOP-SSAR, interfaces identified in Section VII should be presented for each system (review area) in the following categories to facilitate review. Review areas in Section VII of a broad nature, as discussed in item c above, need not use these categories for presentation of interfaces.

1. Power - Requirements for all types of power for safety systems and components (e.g., offsite power to plant during certain conditions, power from plant to site-specific components).

2. Site Parameters - Site design parameters, based on site characteristics (seismic, geological, hydrological, and meteorological), used for the design of safety systems, components, and structures against naturally occurring events.

3. Missiles - Missiles generated by natural phenomena used as the basis for the design of safety systems, components, and structures.

4. Thermal Limitations - Requirements for cooling safety systems and components, including fluid conditions and limitations.

5. Monitoring - Requirements for performance surveillance, testing, and inspection of interfacing safety systems and components.

6. Actuation/Controls - Requirements for actuation of interfacing systems and components and for control of their subsequent operation.

7. Materials - Requirements for materials for safety systems and components.

8. System/Component Arrangement - Location requirements (including inservice inspection and testing) that safety systems and components place on plant arrangement.

9. Radioactive Waste - Release of radioactive material to the environment.

10. Related Service - Requirements for other essential services for safety systems and components (e.g., interfaces from Section V.f below, fire protection, compressed air).

11. Mechanical Interaction Between Systems and Buildings - Requirements for consideration of differential motion, including seismic effects and thermal expansion.

12. Design Criteria - Criteria upon which BOP system designs, or portions thereof, are based.

f. The physical points of interface for fluid systems should be indicated on piping and instrumentation diagrams (P&IDs) and those for electric systems on elementary, schematic, or logic diagrams and on block diagrams to clearly show the line of demarcation between the standard design and the unspecified matching portions of the plant. Each system interfacing point should be uniquely labeled. The selection of specific interface points should be based on the division of design responsibility, not supply responsibility, established between the reactor vendor and architect-engineer (e.g., an interface point should not be established between a component supplied by the reactor vendor and the piping in the same system supplied by the architect-engineer). Safety-related fluid and electric interfaces applicable to each point, as identified in Sections VI and VII, should be listed, consistent with the system-by-system basis established for the definition of interfaces (e.g., the compressed air requirements for all required air-operated valves in the system as a group, the d.c. power requirements for all d.c. instrumentation in the system as a group).

g. All the standard design interfaces should be addressed in a referencing SAR (either a PSAR for a CP application or another SSAR) for the design of an interfacing system, component, or structure; for the design of utility-specific items; or for the determination of standard plant/site compatibility. The specific interfaces used for each interfacing area should be identified. The identification should be presented in the "Design Bases" section of each system description. Identification should

143 307

consist of appropriate references to the interfaces in the referenced SSAR; the specific interfaces should not be rewritten or reprinted. For interfaces to P&IDs and to electric diagrams in a referencing SAR, a similar procedure should be used.

VI. NUCLEAR STEAM SUPPLY SYSTEM INTERFACES

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.4 Water Level (Flood) Design

3.4.3 NSSS Interface

Safety-related NSSS equipment located outside containment that must be protected from flooding.

3.5 Missile Protection

3.5.4 NSSS Interface

1. NSSS equipment located inside or outside containment that potentially could produce missiles, including type (e.g., valve bonnet, studs, stems, thermowells), weight, size, and energy of each missile.

2. Safety-related NSSS equipment located outside containment requiring protection from externally generated missiles (e.g., tornado missiles).

3. Safety-related NSSS equipment located inside or outside containment requiring protection against internally generated missiles.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.3 NSSS Interface

1. Identification of high- and moderate-energy NSSS pipelines inside and outside containment.

2. Safety-related NSSS systems and equipment located inside or outside containment requiring protection from the effects of failures of high- and moderate-energy pipelines.

143 308

3. The coordination of the design of the RCS with interfacing BOP-designed piping systems regarding postulated pipe break locations, orientation, configurations, and resulting loads to ensure compatibility.[1]

3.7 Seismic Design

3.7.5 NSSS Interface

1. A listing of NSSS systems and components that, in conjunction with supporting structures, are designed to Seismic Category I requirements. (Information given in Section 3.2.1 of the SAR may be referenced here.)

2. The establishment at all support points of the seismic response spectra envelopes to which the standard NSSS equipment is designed for use in the BOP design.[1]

3. Envelopes of allowable seismic loads transmitted from Category I or non-Category I systems that connect to the standard NSSS components for use in the BOP design.[1]

4. The mass and stiffness properties of the NSSS to be coupled with the mathematical model of the seismic system including structures and supports, for use in the BOP design.[1]

3.8 Design of Category I Structures

3.8.6 NSSS Interface

1. The maximum differential displacements and rotations due to the loads at points of the NSSS that will interface with BOP structures for use in the BOP design.[1]

2. The range of structural properties of supporting BOP structures that were used in the analysis of the NSSS for use in the BOP design.[1]

3. All the loads that have to be transmitted from the NSSS components to the supporting BOP structures for use in the BOP design.[1]

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[1]This interface involves the exchange of information among the utility, the NSSS designer, and the BOP designer to ensure compatibility of interfacing systems, components, and structures. This information exchange takes place in accordance with the requirements of Appendix B to 10 CFR Part 50. The information need not be provided to the NRC unless specifically requested by the staff; however, the fact that such information was exchanged and the necessary evaluations were performed should be documented in the SSAR.

143 309

4. Evaluation of the deflections under all loading conditions, provided by the BOP designer for the BOP structures supporting NSSS components.

5. Evaluation of design information and drawings prepared by the BOP designer, as they affect NSSS systems and components to ensure compliance with NSSS design criteria.*

3.9 Mechanical Systems and Components

3.9.7 NSSS Interface

1. The coordination of the design of the RCS and interfacing BOP-designed systems, components, and supports when inelastic analysis methods are used by either the NSSS or BOP designer to ensure compatibility. Areas requiring coordination should include analytical criteria, procedures, and results.*

2. Preoperational piping vibration test parameters for the NSSS system and components for all ASME Class 1, 2, and 3 piping systems for use in the BOP design.*

3. The establishment of the test program for the flow-induced vibration of reactor internals for use by the utility. (Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," is applicable reference.)[2]

4. The coordination of the design of NSSS active and inactive components and their supports with the design of interfacing BOP components and supports regarding design loading combinations to ensure structural and functional compatibility. The categorization of the appropriate plant and component operating conditions should be coordinated with the BOP designer.*

5. The coordination of the structural and functional aspects of overpressure protection for NSSS-designed systems and components with the BOP designer to ensure compatibility.*

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[2]The staff recognizes that the information may not be available at the PDA stage of review. If all the information needed by the staff and ACRS to complete their review of this interface is not provided in the application for a PDA, the PDA will be subject to a condition that either the additional information be provided to the utility for inclusion in a CP application referencing the PDA or the utility must demonstrate in its CP application that such information may reasonably be left for later consideration in accordance with §50.35(a) of 10 CFR. Issuance of a PDA does not foreclose staff and ACRS review of interfaces subject to such a condition.

*See footnote 1 on page A-11.

143 310

6. Limiting criteria affecting NSSS active component operability for use in the BOP design.*

7. The development of reference test data for inservice testing of NSSS pumps and valves as specified in subsections IWP and IWV of ASME Section XI for use by the utility.*

3.10 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

3.10.5 NSSS Interface

The coordination of the seismic design requirements of all NSSS safety-related instrumentation and electrical equipment and supports with regard to the floor response spectra defined by the BOP designer to ensure compatibility.*

3.11 Environmental Design of Mechanical and Electrical Equipment

3.11.6 NSSS Interface

1. Heat loads and environmental requirements for NSSS equipment located outside containment.

2. Maximum and minimum containment environmental conditions (i.e., temperature, pressure, humidity, radiation level, etc.) to which NSSS safety-related mechanical and electrical equipment is qualified.

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the design of the RCPB.

2. Quantity of reactor coolant transferred to secondary side of the steam generator following a tube failure; time to effect pressure equalization between a defective steam generator and the RCS; and minimum water volume and maximum steam volume on the secondary side of a steam generator during normal operation.

3. Steam and feedwater conditions (i.e., flow, pressure, and temperature) under all modes of operation, including startup and shutdown.

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*See footnote 1 on page A-11.

143 311

4. Minimum total capacity and maximum set pressures for secondary safety valves (ASME Code Section III), maximum accumulation, division of relief capacity among main steam lines (including maximum flow per valve at set pressure), and minimum or limiting steamflow for atmospheric relief valves for each main steam line, including pressure and temperature.

5. For PWRs only, design requirements for the piping connecting the pressurizer to the pressurizer relief tank (including maximum steam-flow to be accommodated and maximum back pressure at valve discharge).

6. Volume of reactor coolant contained within the RCPB.

7. Requirements for leak detection systems (e.g., type of leakage, locations, rates) to permit control room monitoring of identified and unidentified leakage from the RCPB to containment and of intersystem leakage from the RCPB (including leakage to the secondary side of a steam generator).

8. For BWRs only, mass and energy release rate data for safety and relief valve discharges during anticipated transients.

9. For BWRs only, assumed impulse loads to which NSSS system and components may be subjected due to pool swell forces during a LOCA blowdown. Coordination of these loads with the specific containment design provided by the BOP designer to ensure compatibility.

10. For PWRs only, maximum steam generator mass and energy release rate data for a spectrum of steam and feedwater line break sizes inside containment and selected plant operating conditions; and requirements for isolating flow to any secondary system pipe break, including MSLIV's closure time.

11. For PWRs only, mass and energy release rate data for selected RCS break sizes and locations and the assumed maximum containment design pressure used to generate the mass and energy release rate data. Coordination of the assumed pressure with the containment analysis performed by the BOP designer to ensure that the maximum containment pressure calculated by the BOP designer does not exceed the pressure assumed by the reactor vendor.*

12. For BWRs only, mass and energy release rate data for a spectrum of assumed main steam line and recirculation line break sizes of selected RCS piping locations; mass and energy release rate data for RHRS head spray line and RWCUS line breaks.

13. For PWRs only, requirements for auxiliary feedwater provisions to all intact steam generators, with or without offsite and normal onsite power available, assuming isolation of a steam generator, if applicable, due to a pipe break event such as a steam line break, feedwater line break, or a steam generator tube rupture:

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*See footnote 2 on page A-12.

143 312

a. Reliability requirements (i.e., redundancy, diversity, etc.),

b. Minimum flow capability and maximum flow to a single steam generator,

c. Minimum discharge pressure,

d. Maximum time to attain full flow following demand signal,

e. Minimum volume of stored condensate or standby auxiliary feedwater required to bring reactor to cold shutdown with zero time at hot standby and no allowance to maintain reactor at cold shutdown,

f. Temperature limits,

g. Quantity and condition of steam available from intact steam generator for motive power (final values to be coordinated with the BOP designer to ensure compatibility),

h. Conditions within the NSSS that initiate flow,

i. Conditions within the BOP that are assumed to initiate flow, and

j. Requirements for compatibility of the control and power systems with the NSSS actuation and redundancy logic.

14. Maximum stroke time for MSLIV; requirements for the location, capacity, and control arrangement of the main steam line relief and dump valves and other main steam system valves located between the MSLIV and the main condenser; and requirements for the feedwater control system.

15. Limiting heat loads and coolant conditions (flow, pressure, and temperature) for the condensate storage facilities for all plant modes of operation, including accident conditions; and minimum water inventory for cold shutdown.

16. For PWRs only, temperature, pressure, radioactivity concentrations, and flow rate to the steam generator blowdown system during normal and anticipated operational occurrences; and isolation requirements.

17. For PWRs only, requirements to maintain secondary side water chemistry for steam generators within specified ranges (including steam generator blowdown, chemical addition, condensate purification, and monitoring).

18. Requirements to provide capability for sampling to monitor fluid system performance (including instrumentation for monitoring impurity

143 313

removal and for detecting excessive chloride and fluoride content). For PWRs only, requirements to sample and analyze reactor coolant for specified parameters at hot leg, pressurizer surge line, and pressurizer steam space. (Regulatory Guides 1.44, "Control of the Use of Sensitized Stainless Steel," for PWRs and 1.56, "Maintenance of Water Purity in Boiling Water Reactors," for BWRs are applicable references.)

19. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

20. Reactor coolant radioactivity concentrations, noble gas release rates (BWRs only), and leak rates from RCS to floor drains and building atmospheres.

21. Flow rate, batch volume, radioactivity concentrations, temperature, pressure, and partition factors at each RCS interface point with the process sampling system and for each leakage point to the building atmosphere during normal and anticipated operational occurrences.

22. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each RCS component interfacing with the SSWS or the CCWS.

23. Materials Interfaces 1 to 6 and 8, Table 1.

24. Locations and accessibility requirements for inservice inspection of ASME Code Class 1 components within RCPB. (ASME Code Section XI is applicable reference.)

25. Locations and accessibility requirements for inservice inspection of reactor coolant pump flywheels. (Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," is applicable reference.)

26. Locations and accessibility requirements for inservice inspection of steam generator tubes. (Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," is applicable reference.)

27. Criteria for contamination protection and cleaning before, during, and after welding installation of steam generators at NSSS-BOP boundaries (to avoid stress-corrosion cracking of Inconel tubes). (Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," is applicable reference.)

28. To preclude adverse effects on NSSS equipment, compatibility requirements for materials to be used in containment spray system, considering reactor coolant and radiation environment during accident conditions. (Austenitic stainless steel not sensitized or alternative steel specified

Table 1

MATERIALS INTERFACES COMMON TO SEVERAL SYSTEMS

1. Criteria for contamination protection and cleaning before, during, and after field welding installation of austenitic stainless steel components at NSSS-BOP boundaries.* (Regulatory Guides 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and 1.44, "Control of the Use of Sensitized Stainless Steel," are applicable references.)

2. Requirements for control of sensitization of field installation welds joining austenitic stainless steel components at NSSS-BOP boundaries. (Regulatory Guide 1.44 is applicable reference.)

3. Requirements for control of delta ferrite in field installation welds joining austenitic stainless steel components at NSSS-BOP boundaries. (Regulatory Guides 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and 1.44 are applicable references.)

4. Welding requirements for field welding installation of ferritic steel components and austenitic stainless steel components at NSSS-BOP boundaries, including preheat temperature control, welding materials, and clad welding requirements. (Regulatory Guides 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," and 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," are applicable references.)

5. Requirements for low halide nonmetallic thermal insulation on austenitic stainless steel at NSSS-BOP boundaries. (Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," is applicable reference.)

6. Material requirements for BOP piping connected to all fluid systems of the NSSS, including RCS. (Regulatory Guide 1.44 is applicable reference.)

7. Requirements to provide capability for sampling to monitor fluid system performance.

8. Requirements to provide capability for fluid purity and chemistry control within specified ranges during operation. (Regulatory Guides 1.44 for PWRs and 1.56, "Maintenance of Water Purity in Boiling Water Reactors," for BWRs are applicable references.)

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*NSSS-BOP boundary is a boundary between an NSSS system and a BOP system or a boundary between an NSSS component and a BOP component within an NSSS system.

143 315

by NSSS designer for metals contacting reactor coolant and materials resistant to environment.) (Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," is applicable reference.)

29. Heat loads and environmental limitations for NSSS systems and components.

30. Reactor coolant pump trip signals initiated by main turbine stop valve closure.

5.4 Component and Subsystem Design

5.4.6 Reactor Core Isolation Cooling System

5.4.6.5 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RCICS included within the NSSS.

2. Net positive suction head requirements at RCICS pump suction and required heat removal capacity (including tube-side coolant conditions) during all conditions of standby and shutdown cooling until reactor vessel is depressurized.

3. Mass and energy release rates for RCICS line breaks.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each RCICS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each RCICS component interfacing with the SSWS.

6. Materials Interfaces 1 to 8, Table 1.

7. Requirements for safety and relief valve back pressure, spatial separation, and discharge piping.

5.4.7 Residual Heat Removal System

5.4.7.5 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RHRS included within the NSSS.

143 316

2. Net positive suction head requirements at the RHRS pump suction and required heat removal capacity (including tube-side coolant conditions) during all conditions of shutdown cooling.

3. For BWRs only, mass and energy release rates for RHRS head spray line breaks.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each RHRS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each RHRS component interfacing with the CCWS or SSWS.

6. Materials Interfaces 1 to 8, Table 1.

7. Requirements for safety and relief valve back pressure, spatial separation, and discharge piping.

5.4.8 Reactor Water Cleanup System (BWRs)

5.4.8.4 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RWCUS included within the NSSS.

2. Mass and energy release rates for RWCUS line breaks.

3. Flow rate, batch volume, radioactivity content, and batch frequency for filters, filter sludges, demineralizer resins, and evaporator bottoms transferred from RWCUS equipment to the SWMS during normal and anticipated operational occurrences.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each RWCUS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Heat loads and service water flow, pressure, and temperature for normal and limiting conditions for each RWCUS component interfacing with the SSWS.

6. Materials Interfaces 1 to 8, Table 1.

143 3'

5.4.11 Pressurizer Relief Discharge System (PWR)

5.4.11.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the PRDS included within the NSSS.

2. Heat loads and cooling water flow, pressure, and temperature for normal and limiting conditions for each PRDS component interfacing with the CCWS.

3. Materials Interfaces 1 to 8, Table 1.

6. ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.4 Containment Isolation System

6.2.4.5 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the CIS included within the NSSS.

2. Definition of signals generated by NSSS equipment (i.e., safety injection, low vessel level, etc.) for use in developing diverse containment isolation signals, and characteristics of valves included in the NSSS design that are part of the CIS. Definition of BOP signals that indicate abnormal containment conditions and that must be coordinated with the NSSS in the design of the reactor shutdown systems to ensure compatibility.

3. Maximum leakage rate and type of fluid for all containment isolation devices included in the NSSS design.

4. Test fluid type and maximum quantity of test fluid required for testing of containment isolation devices included in the NSSS design.

5. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

6.3 Emergency Core Cooling System

6.3.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the ECCS included within the NSSS.

143 318

2. Maximum head loss (friction and elevation), minimum net positive suction head requirements for all ECCS pumps for all conditions of operation (including single failure, operator error, minimum containment ambient pressure, and long- and short-term cooling), and identification of all conditions under which ECCS must provide core cooling (e.g., single failure, flooding); consider recirculation mode using containment sump or suppression pool.

3. For PWRs only, design requirements for the piping connecting the accumulators to the RCS and provisions for nitrogen supply.

4. For BWRs only, design requirements for the piping connecting the ADS accumulator to the relief valves and provisions for air supply.

5. Design requirements for manually operated valves in the ECCS; requirements for straight piping runs for flow measuring devices in ECCS; limitations on total water volume in the RCS cold leg up to the ECCS check valves; maximum time to achieve full ECCS flow in the event of a LOCA (with and without the availability of normal a.c. power supply); limitations on particle size of impurities in ECCS water; requirements for venting and filling provisions for air removal to preclude water hammer events; and design capability for preoperational testing to demonstrate all aspects of system operability.

6. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

7. Hydrogen released by zirconium water reaction in core; maximum amount of hydrogen dissolved in RCS water during plant operations (PWRs only); hydrogen generated by radiolysis of water in the reactor and in the containment sump as a function of time after LOCA; surface area, weight, and thickness of aluminum and zinc provided as a part of NSSS equipment inside containment; and hydrogen generation rate due to corrosion of aluminum and zinc by containment spray post-LOCA. (Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," is applicable reference.)

8. For PWRs only, assumed containment parameters are:

a. Maximum passive heat sinks (materials of construction, surface area, thickness),

b. Maximum free volume in containment,

c. Containment initial conditions (temperature, pressure, and humidity), and

d. Maximum containment active heat removal capability (heat removal rates, start times, containment spray flow rate and temperature, etc.).

Minimum containment pressure analysis must be coordinated with the BOP designer using actual containment parameters to ensure compatibility.

9. For BWRs only, minimum containment pressure assumed is 14.7 psia; actual pressure must be determined in coordination with the BOP designer using actual containment parameters to ensure compatibility.

10. Requirements on recirculation water pH for emergency core cooling and containment cooling.

11. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each ECCS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

12. Heat loads, cooling water flow, pressure, and temperature for normal and limiting conditions for each ECCS component interfacing with the SSWS or the CCWS.

13. Materials Interfaces 1 to 8, Table 1.

14. Requirements for safety and relief valve back pressure, spatial separation, and discharge piping.

6.4 Habitability Systems

6.4.7 NSSS Interface

1. Safety-related NSSS control equipment located in the control room.

2. Limiting design and operational requirements of NSSS control equipment (e.g., temperature, humidity).

6.6 Inservice Inspection of Class 2 and 3 Components

6.6.9 NSSS Interface

Locations and accessibility requirements for inservice inspection of all ASME Code Class 2 and 3 components within the NSSS auxiliary systems and ESFs. (ASME Code Section XI is applicable reference.)

6.7 Main Steam Line Isolation Valve Leakage Control System (BWRs)

6.7.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the MSIVLCS included within the NSSS.

143 320

2. MSIVLCS parameters, including MSLIV leak rate, concentrations of radioactivity in steam, and setpoints on instrumentation, to be interlocked with leakage control system. (Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," is applicable reference.)

3. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

4. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each MSIVLCS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

5. Materials Interfaces 1 to 8, Table 1.

7. INSTRUMENTATION AND CONTROLS

7.2 Reactor Trip System

7.2.3 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the RTS included within the NSSS.

2. Requirements for anticipatory trips (e.g., turbine trip signals as input to the reactor trip system).

7.8 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the I&C system included within the NSSS.

2. For each NSSS system, requirements for NSSS instrumentation energized by the plant instrumentation power supply system:

a. Steady-state load,

b. Transient/step load,

c. Nominal system voltage,

d. Allowable voltage regulation, 143 321

e. Allowable harmonic content,

f. Allowable frequency fluctuation,

g. Grounding requirements,

h. Power supply assignment, and

i. Percent peak deviation from true sine wave (inverters).

3. For each NSSS system, requirements for BOP sensors that provide inputs to accomplish NSSS functions and for associated instrument lines:

a. Range (including accident conditions),

b. Measurement accuracy,

c. Repeatable accuracy,

d. Maximum expected transient,

e. Response time (maximum allowable time to achieve sensor output after reaching trip level for measured variable),

f. Trip setpoint,

g. Snubbers,

h. Orifice,

i. Arrangement for instrument lines,

j. Type and location of readout, and

k. Bypass and inoperable status indication.

4. Number of logic trains used for the control of safety systems.

8. ELECTRIC POWER

8.2 Offsite Power System

8.2.3 NSSS Interface

For each NSSS system, requirements for offsite a.c. power:

1. Steady-state load,

2. Inrush kVA for motor loads,

3. Nominal voltage,

4. Allowable voltage [illegible]ation,

5. Nominal frequency,

6. Allowable frequency fluctuation,

7. Maximum frequency decay rate and limiting underfrequency value for reactor coolant pump coastdown, and

8. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

8.3 Onsite Power Systems

8.3.1 A.C. Power Systems

8.3.1.5 NSSS Interface. For each NSSS system, requirements for onsite a.c. power:

1. Steady-state load,

2. Inrush kVA for motor loads,

3. Nominal voltage,

4. Allowable voltage drop (to achieve full functional capability within required time period),

5. Sequence and time to achieve full functional capability for each load,

6. Nominal frequency,

7. Allowable frequency fluctuation,

8. Number of trains, and

9. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

8.3.2 D.C. Power Systems

8.3.2.3 NSSS Interface. For each NSSS system, requirements for onsite d.c. power:

1. Steady-state load,

2. Surge loads (including emergency conditions),

3. Load sequence,

4. Nominal voltage,

5. Allowable voltage drop (to achieve full functional capability within required time period),

143 323

6. Number of trains, and

7. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

9. AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.4 NSSS Interface. Rack dimensions, weight, materials of construction, uplift forces, and mounting requirements; minimum storage capacity; minimum rack spacing and associated k_{eff} (with flooded unborated water and with optimum moderator aqueous foam); and vault drainage requirements.

9.1.2 Spent Fuel Storage

9.1.2.4 NSSS Interface

1. Rack dimensions, weight, materials of construction, uplift forces, and mounting requirements; minimum storage capacity; minimum rack spacing and associated k_{eff} (borated and unborated water); and allowable fuel pool water chemistry (e.g., pH, conductivity, boron concentrations) and limiting water temperature.

2. Minimum depth of water above spent fuel array to meet shielding requirements of 10 CFR Part 20; minimum depth of water above spent fuel bundle if accidentally dropped and positioned horizontally across top of spent fuel array; spent fuel pool normal and maximum decay heat loads (including fraction of core and minimum cooldown time prior to placing in the pool); and potential corrosion rate of racks and cladding and expected fission product leakage as a function of temperature and water chemistry.

9.1.4 Fuel Handling System*

9.1.4.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the Fuel Handling System included within the NSSS.

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*Fuel Handling System assumed to consist of the following NSSS items: bridge cranes; new fuel elevator; transfer tube and carriage; upender; and lifting rigs, slings, and other essential equipment integral to the NSSS.

2. Installation requirements (i.e., crane weights, power, compressed air, hydraulic requirements, etc.); storage requirements; capacity of bridge cranes and new fuel elevator; installation requirements for transfer tube, carriage, and upender; equipment interlocks and special built-in safety features; and other special requirements to preclude unacceptable accidents.

9.3 Process Auxiliaries

9.3.4 Chemical and Volume Control System (PWRs)

9.3.4.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the CVCS included within the NSSS.

2. Location of the CVCS letdown line radiation monitor and requirements to perform its alarm and control function.

3. Flow rate, batch volume, radioactivity content, and batch frequency for filters, filter sludges, demineralizer resins, and evaporator bottoms transferred from the CVCS equipment to the SWMS during normal and anticipated operational occurrences.

4. Flow rate, batch volume, radioactivity concentrations, temperature, pressure, and partition factors at each CVCS interface point with the GWMS and for each leakage point to the building atmosphere during normal and anticipated operational occurrences.

5. Flow rate, batch volume, radioactivity concentrations, temperature, and pressure at each CVCS interface point with the LWMS and for each leakage point to the building sump during normal and anticipated operational occurrences.

6. Heat loads and cooling waterflow, pressure, and temperature for normal and limiting conditions for each safety-related CVCS component interfacing with the CCWS.

7. Flow rate, boron concentrations, temperature, and pressure at each CVCS interface point with the refueling water or borated water storage tank and volume available in excess of requirements for accident condition.

8. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

9. Materials Interfaces 1 to 8, Table 1.

9.3.5 Standby Liquid Control System (BWRs)

9.3.5.6 NSSS Interface

1. A listing of all design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the SLCS included within the NSSS.

2. Boron concentration, flow rate, and requirements for maintaining minimum temperature.

3. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

4. Materials Interfaces 1 to 8, Table 1.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

9.5.1.6 NSSS Interface. Identification, quantification, and tabulation of the NSSS items that constitute a significant fire hazard.

10. STEAM AND POWER CONVERSION SYSTEM

10.4 Other Features of Steam and Power Conversion System

10.4.4 Turbine Bypass System

10.4.4.X NSSS Interface

1. Steam conditions during discharge following turbine trip and limiting steamflow for turbine bypass system sizing without reactor trip, including temperature and pressure.

2. Identification of valves, instruments, and controls essential to safe shutdown of the plant that may be pneumatically operated; include airflow, pressure, cleanliness, and dew point requirements.

13. CONDUCT OF OPERATIONS

13.6 Industrial Security

13.6.3 NSSS Interface*

Identification of vital equipment as defined in 10 CFR §73.2 for use in developing physical security plans. Also, where applicable, locations

*This information should be submitted as proprietary information [10 CFR §2.790(d)].

143 326

of items of vital equipment and provisions incorporated into the design for monitoring the status of vital equipment to detect malevolent acts to impair performance.

14. INITIAL TEST PROGRAM

14.1 Specific Information To Be Included in Preliminary Safety Analysis Reports

14.1.8 NSSS Interface

Identification of the special or unique features of the NSSS design for consideration in developing the initial test program by the utility.

15. ACCIDENT ANALYSES

15.X Evaluation of Individual Initiating Events

15.X.X Event Evaluation

15.X.X.X NSSS Interface. Identification of the Safety Actions required to mitigate the consequences of each transient and accident event and the BOP system necessary to provide each Safety Action.

VII. BALANCE-OF-PLANT INTERFACES

2. SITE CHARACTERISTICS

2.3 Meteorology

2.3.6 BOP Interface

Limiting meteorological parameters (χ/Q) for design basis accidents (including use of backup hydrogen purge system) and for routine releases and other extreme meteorological conditions (e.g., temperatures, winds, humidity, dust storms, air quality, and appropriate combinations) for the design of systems and components exposed to the environment.

3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.3 Wind and Tornado Loadings

3.3.3 BOP Interface

Tornado and operating basis wind loadings to which plant structures and exposed systems and components are designed.

143 327

3.4 Water Level (Flood) Design

3.4.3 BOP Interface

1. Flood and ground water elevation and requirements for protection established for the design of systems, components, and structures.

2. Hydrostatic loads established for the design of systems, components, and structures.

3.5 Missile Protection

3.5.4 BOP Interface

1. External missiles generated by natural phenomena established for the design of systems, components, and structures.

2. External missiles resulting from man-made hazards and accidents.

3.7 Seismic Design

3.7.5 BOP Interface

Seismic parameters ("g" values and response spectra) established for the design of systems, components, and structures. In addition, the range of site parameters (e.g., shear wave velocity, depth of embedment, depth of overburden) that are used in the seismic analysis should be specified.

3.8 Design of Category I Structures

3.8.6 BOP Interface

1. Snow, ice, and rain loads established for the design of systems, components, and structures.

2. Required bearing capacity of foundation materials, allowable absolute and differential settlements, and allowable tilt established for the design of systems, components, and structures.

3. Lateral earth pressure loads established for the design of systems, components, and structures.

4. Maximum loads, including overpressurization resulting from man-made hazards and accidents such as potential explosions and associated missiles in vicinity of the plant, potential aircraft impacts, etc.

143 328

3.9 Mechanical Systems and Components

3.9.7 BOP Interface

1. The preoperational piping vibration test parameters for use by the utility in developing the test programs for all ASME Class 1, 2, and 3 piping systems in the NSSS and BOP, including all high-energy piping systems outside containment and all Seismic Category I portions of moderate-energy piping systems outside containment.

2. The locations and other requirements for use by the utility in developing an inservice inspection program for ASME Class 1, 2, and 3 systems and components in the NSSS and BOP and for reactor coolant pump flywheels (PWRs only).*

3. Reference test data for inservice testing of BOP pumps and valves as specified in subsections IWP and IWV of ASME Section XI for use by the utility.

6. ENGINEERED SAFETY FEATURES

6.4 Habitability Systems

6.4.7 BOP Interface

Environmental conditions assumed in the design of the control room for protection of operators.

7. INSTRUMENTATION AND CONTROLS

7.8 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the I&C systems included within the BOP.

2. Provisions included in the plant instrumentation power supply system to accommodate the I&C requirements of the SSWS (and additional cooling capacity, if any required):

   a. Steady-state load,

   b. Transient/step load,

   c. Nominal system voltage,

   d. Allowable voltage regulation,

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*See footnote 2 on page A-12.

e. Allowable harmonic content,

f. Allowable frequency fluctuation,

g. Grounding requirements, and

h. Power supply assignment.

3. Provisions included for the sensors and their instrument lines associated with the SSWS (and additional cooling capacity, if any required) that provide inputs to satisfy station safety functions:

a. Range (including accident conditions),

b. Measurement accuracy,

c. Repeatable accuracy,

d. Maximum expected transient,

e. Response time (maximum allowable time to achieve sensor output after reaching trip level for measured variable),

f. Trip setpoint,

g. Snubbers,

h. Orifice, and

i. Arrangement for instrument lines.

8. ELECTRIC POWER

8.2 Offsite Power System

8.2.3 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the offsite power system included within the BOP.

2. For each BOP system, requirements for offsite a.c. power system:

a. Steady-state load,

b. Inrush kVA for motor loads,

c. Nominal voltage,

d. Allowable voltage regulation,

e. Nominal frequency,

f. Allowable frequency fluctuation,

g. Maximum frequency decay rate and limiting underfrequency value for reactor coolant pump coastdown, and

h. Minimum number of ESF trains to be energized simultaneously (if more than two trains provided).

NOTE: For complete offsite a.c. power requirements, the BOP designer should also include the NSSS requirements.

8.3 Onsite Power System

8.3.1 A.C. Power Systems

8.3.1.5 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the onsite a.c. power systems included within the BOP.

2. For each BOP system, requirements for onsite a.c. power:

a. Steady-state load,

b. Inrush kVA for motor loads,

c. Nominal voltage,

d. Allowable voltage drop (to achieve full functional capability within required time period),

e. Load sequence,

f. Nominal frequency, and

g. Allowable frequency fluctuation.

NOTE A: For complete onsite a.c. power requirements, the BOP designer should also include the NSSS requirements.

NOTE B: This interface assumes that the onsite a.c. diesel generator system is utility-specific.

3. Coordination of the design of the diesel generator room with the utility-applicant.

143 331

8.3.2 D.C. Power Systems

8.3.2.3 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the d.c. power systems included within the BOP.

2. Provisions included to accommodate the needs of the SSWS (and additional cooling capacity, if required) by the d.c. power systems:

a. Steady-state load,

b. Surge loads,

c. Load sequence,

d. Nominal voltage, and

e. Allowable voltage drop (to achieve full functional capability within required time period).

9. AUXILIARY SYSTEMS

9.2 Water Systems

9.2.7 BOP Interface

1. A listing of all the design criteria including codes, standards, General Design Criteria, and regulatory guides applied to the portion of the design of the SSWS included within the BOP.

2. Integrated heat load (decay heat and station heat load for all NSSS and BOP systems, as a function of time for the various modes of plant operation and limiting accident conditions) that must be transferred to the ultimate heat sink, maximum and minimum temperature limits, pressure, flow rate, plant SSWS pressure drop, etc.

3. Coolant flow, pressure, temperature, and integrated condensate storage capacity to satisfy total plant needs during normal operation, shutdown, and accident conditions. Cooling water requirements for the diesel generator system should be coordinated with the utility-applicant.

4. Limits on quality of makeup water to the station, including conductivity, pH, oxygen, chlorides, fluorides, solids, carbon dioxide, particulates, and silica; and limits on makeup waterflow, temperature, and pressure.

5. Requirements for location and arrangement of potable and sanitary water systems to preclude adverse effects on safety systems and components in the event of failure.

143 332

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.6 BOP Interface

Ventilation requirements for the diesel generator rooms should be coordinated with the utility-applicant.

9.5 Other Auxiliary Systems

9.5.9 BOP Interface

Site-related requirements to satisfy the fire protection program.

11. RADIOACTIVE WASTE MANAGEMENT

11.2 Liquid Waste Management Systems

11.2.4 BOP Interface

Expected release rates of radioactive material to the environment from the LWMS.

11.3 Gaseous Waste Management Systems

11.3.4 BOP Interface

Expected release rates of radioactive materials to the environment from the GWMS and from other release points including:

1. Location of all release points,
2. Height above grade,
3. Height relative to adjacent buildings,
4. Effluent temperature,
5. Effluent flow rate,
6. Effluent velocity, and
7. Size and shape of flow orifice.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.3 Effluent Monitoring and Sampling

11.5.3.X BOP Interface. Requirements for offsite sampling and monitoring of effluent concentrations.

143 333

13. CONDUCT OF OPERATIONS

13.3 Emergency Planning

13.3.3 BOP Interface

Features of the plant that may affect plans for coping with emergencies, as specified in Appendix 0 to 10 CFR Part 50. Examples of such features are the design of the onsite emergency first aid and personnel decontamination facilities and the emergency operations center, and facility features, including communications systems, that ensure the capability for plant evacuation and reentry (i.e., to mitigate the consequences of an accident or, if appropriate, to continue operations).

13.6 Industrial Security

13.6.3 BOP Interface*

Identification of vital equipment, as defined in 10 CFR §73.2, for use in developing physical security plans. Also, where applicable, locations of items of vital equipment and provisions incorporated into the design for monitoring the status of vital equipment to detect malevolent acts to impair performance.

14. INITIAL TEST PROGRAM

14.1 Specific Information To Be Included in Preliminary Safety Analysis Reports

14.1.8 BOP Interface

Identification of the special or unique features of the NSSS and BOP designs for consideration in developing the initial test program by the utility.

*This information should be submitted as proprietary information [10 CFR §2.790 (d)].

143 334

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