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REVISION 3

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

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Appendix A INTERFACES FOR STANDARD DESIGNS

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).

Revisions

Data and text should be updated or revised by replacing pages. "Pen and ink" or "cut and paste" changes should not 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 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.

5. <u>Printing</u>

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. <u>Binding</u>

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

7. <u>Page Numbering</u>

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.)

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.

* 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,

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.

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 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.

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.

*"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 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 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.

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.

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 2.2.2, the potential accidents to be considered as design basis events should be determined and the potential effects of these accidents on the nuclear

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

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

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.

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-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.

<u>2.3.2.2 Potential Influence of the Plant and Its Facilities on Local Meteorology</u>. 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 from the plant to a distance of 50 miles (80 km).

<u>2.3.2.3 Local Meteorological Conditions for Design and Operating Bases</u>. 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 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.

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) 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.

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 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.

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.

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 of phreatic 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 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 recorded data, and (8) plans for alarm devices to ensure sufficient time for initiation of corrective action.

j. 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 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 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 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, 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 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 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,

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.

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,

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, 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 I 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.

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,

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 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.

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.

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.

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.

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 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:

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 $\pm$ 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 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 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 systems 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 description 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).

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.

<u>3.7.1.2 Design Time History</u>. 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.

<u>3.7.1.3 Critical Damping Values</u>. 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.

<u>3.7.1.4 Supporting Media for Seismic Category I Structures</u>. 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 <u>Seismic System Analysis</u>

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.

<u>3.7.2.1 Seismic Analysis Methods</u>. 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

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 motion in determining the seismic response of structures, systems, and components follow the recommendations of Regulatory Guide 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 description of the procedure for combining modal responses (shears, moments, stresses, deflections, and accelerations) should be provided. Indicate the extent to which the recommendations of Regulatory Guide 1.92 are followed.

3.7.2.8 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.

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.

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.

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.

<u>3.8.1.4 Design and Analysis Procedures</u>. 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 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.

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.

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.

3.8.2.2 Applicable Codes, Standards, and Specifications. 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.

3.8.2.3 Loads and Load Combinations. 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.

3.8.2.4 Design and Analysis Procedures. 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.

3.8.2.5 Structural Acceptance Criteria. 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.

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.

* 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.

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.

<u>3.8.3.4 Design and Analysis Procedures</u>. 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.

3.8.3.5 Structural Acceptance Criteria. 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.

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.

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

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 as appropriate.

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 each 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.

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 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.

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions. 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.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals. 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 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.

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:

1. A summary description of any test models used (see Section 3.9.1.3).

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 effects 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 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.

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).

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 basis 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.

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 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 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,

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,

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.

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 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.

4.3.2.4 Control Requirements. 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.

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.

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,

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 components 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 temperature 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.

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.

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.

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.

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.

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 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.

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.

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 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.

5.4.2.2 Steam Generator Inservice Inspection. 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.

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

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,

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.

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.

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.

5.4.12 Valves

5.4.13 Safety and Relief Valves

5.4.14 Component Supports