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

Prepared by the Regulatory Staff
U.S. Atomic Energy Commission
Issued February, 1972

FOREWORD

Section 50.34 of 10 CFR Part 50 of the regulations of the Atomic Energy Commission requires that each application for a construction permit for a nuclear reactor facility shall include a preliminary safety analysis report (PSAR), and that each application for a license to operate such a facility shall 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 AEC on June 30, 1966.

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

In 1970, the Commission began issuance of a series of Safety Guides to inform applicants of solutions to specific safety issues that are 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.

On November 18, 1971, the AEC Director of Regulation announced* that effective immediately 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. This document provides a standard format for safety analysis reports and identifies the principal information needed. It supersedes the guide issued in 1966.

Safety Analysis Reports will be expected to conform to this Standard Format unless there is good reason for not doing so. This Standard Format incorporates two Information Guides previously issued, and other information that was being developed for issuance as Information Guides. In the future, the Information Guide Series will be used to publish any revisions or additions to the contents of this Standard Format.

*AEC Press Release No. S-25-71

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

PAGE NO.

CHAPTER 5.0 REACTOR COOLANT SYSTEM

CHAPTER 6.0 - ENGINEERED SAFETY FEATURES

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

PAGE NO.

CHAPTER 10.0 - STEAM AND POWER CONVERSION SYSTEM

CHAPTER 11.0 - RADIOACTIVE WASTE MANAGEMENT

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

INTRODUCTION

Purpose and Applicability

This document has been prepared by the AEC regulatory staff to provide a standard format for Safety Analysis Reports submitted as part of applications for construction permits and operating licenses for nuclear power plants, and to indicate the information to be provided in the reports. The principal purpose for the preparation and submittal of a Safety Analysis Report (SAR) is to inform the Commission of the nature of the facility and plans for its use. The information provided in the SAR must be sufficient to permit a review of whether the facility can be built and operated without undue risk to the health and safety of the public. An applicant will have evaluated the facility in sufficient detail to conclude that it can be built and operated safely. The Safety Analysis Report is the principal document whereby the applicant provides the information needed to understand the basis upon which this conclusion has been reached.

The required content of a Safety Analysis Report is described in general terms in Section 50.34 of the Commission's regulations (10 CFR Part 50). The Standard Format identifies the principal detailed information that is required by the staff in its evaluation of the application. This format will help assure the completeness of the information provided, will assist the regulatory staff and others in locating the information, and will aid in shortening the time needed for the review process. The Standard Format and Content applies to both a Preliminary Safety Analysis Report (PSAR) and a Final Safety Analysis Report (FSAR), but where specific items of information apply to only one of these reports, it is so indicated in the text.

Although the specific information identified in the Standard Format and Content has been prepared with reference to water-cooled power reactors, the general content and format for the presentation of information is also applicable to power reactors of other types.

The information indicated in the Standard Format and Contents is a minimum for Safety Analysis Reports. It is recognized that all the information that may be required to complete the staff review (or all the information that has been presented in previous SARs) is not identified explicitly, and the applicant should include additional information in the SAR, as appropriate.

Upon receipt of an application, the regulatory staff will perform a preliminary review to determine whether the SAR provides a reasonably complete presentation of the information identified in the Standard Format and Content. If not, further review of the application will not be initiated until a reasonably complete report 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 AEC regulations and guides, the results of recent research and development in nuclear reactor safety, and experience in the construction and operation of nuclear power plants.

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 Preliminary Safety Analysis Report, because the design has not progressed sufficiently at the time of writing, it is not sufficient to note merely that the information is "to be supplied later." The report should state the bases or criteria being used to develop the required information, the concepts and/or alternatives under consideration, and the schedule for completion of the design and submission of the missing information. In general, the Final Safety Analysis Report should describe the final design of the plant.

Use of Standard Format

In the Standard Format, the SAR is divided into seventeen chapters (e.g., Chapter 2.0 Site Characteristics). Within the chapters the material is arranged in sections (e.g., 2.4 Hydrology), subsections (e.g., 2.4.2 Floods), and further subdivisions.

The SAR should follow the numbering system of the Standard Format at least down to the level of subsections. For example, subsection 2.4.2 of the SAR should provide all the information requested within subsection 2.4.2 of the Standard Format.

It is recognized that in many cases the applicant may wish to include appendices to the SAR to provide supplemental information not explicitly identified in the Standard Format. Some examples of such information are: (1) summaries of the manner in which the applicant has treated matters addressed in AEC Safety Guides, or proposed regulations; and (2) supplementary information regarding calculational methods or design approaches used by the applicant or his agents.

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.

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

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

Drawings, maps, diagrams, sketches, and charts should be employed whenever the information can be presented more adequately or conveniently by such means. Due concern should be taken to assure 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 chapter in which they are referenced. In cases where proprietary documents are referenced, a non-proprietary summary description 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).

The assembly of pages of the SAR should be accomplished in a manner permitting the easy insertion of additional pages. For example, pages should be numbered by Chapters rather than sequentially throughout the report, as is done in Standard Format. When the SAR consists of more than one volume, the complete table of contents (for all volumes) should be included in the front of each volume.

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

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

The first chapter of the Safety Analysis Report should present an introduction and general plant description. This chapter should enable the reader to obtain an overall understanding of the facility without having to delve into the subsequent chapters. Review of the detailed chapters which follow can then be accomplished with better perspective and with recognition of the relative safety importance of each individual item to the overall facility design.

1.1 Introduction

This section should present briefly the principal aspects of the overall application. For example, the specific information that should be included is as follows: 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 facility. The facility 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, instrumentation, control and electrical systems, power conversion system, fuel handling and storage systems, 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 facility. 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 the technology as 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 which is used in some safety evaluations.

1.3 Comparison Tables

1.3.1 Comparisons with Similar Facility Designs

This subsection should provide a comprehensive indication of the principal similarities to other power reactor facilities (preferably previously designed or built power reactor facilities or designs) and principal differences from such power reactor facilities. This information should be provided in tabular form, cross-referencing the appropriate 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 facility such as the engineered safety features, the containment concept, instrumentation and electrical systems, the radioactive waste management system, and other principal systems.

1.3.2 Comparison of Final and Preliminary Designs

In a Final Safety Analysis Report (FSAR) tables should be provided to identify clearly all the significant changes that have been made in the facility design since submittal of the Preliminary Safety Analysis Report (PSAR). Each item should be cross-referenced to the appropriate 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 reactor facility. 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 designer, architect-engineer, constructor and plant operator should be delineated.

1.5 Requirements for Further Technical Information

In accordance with Section 50.35 of 10 CFR Part 50, 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 research and development programs that will be required to determine the adequacy of the 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, (4) provide a schedule of completion of the program as related to the projected startup date of the proposed facility, 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 AEC; however, if such references are made, the applicability of each research and development item to the applicant's facility should be discussed.

In the Final Safety Analysis Report this section should include a resume of special research and development programs undertaken to establish the final design and/or to demonstrate the added 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" which 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 or architect-engineers and filed separately with the AEC in support of this application or of other applications or product lines. This tabulation should include for each report the title, the report number, the date submitted to the AEC and the applicable sections of the SAR in which this report is referenced. For any reports that have been withheld from public disclosure, pursuant to Section 2.790(b) of 10 CFR Part 2, as proprietary documents, non-proprietary 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 AEC in other applications that are incorporated in whole or in part in this application by reference.

2.0 SITE CHARACTERISTICS

This chapter of the Safety Analysis Report should provide information on the geological, seismological, hydrological, and meteorological characteristics of the site and vicinity, in conjunction with population distribution, 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

The site location should be described by specifying the latitude and longitude of the reactor to the nearest second, and the Universal Transverse Mercator coordinates* to the nearest 100 meters. The state and county in which the site is located should be identified, as well as the location of the site relative to prominent natural and man-made features such as rivers and lakes.

2.1.2 Site Description

A map of the site should be included in the application and should be of suitable scale to clearly define the boundary of the site and the distance from significant facility features to the site boundary. The area to be considered as the exclusion area must be delineated clearly, if its boundaries are not the same as the boundaries of the plant site. The application should include a description of the applicant's legal rights with respect to the properties described (ownership, lease, easements, etc.).

2.1.2.1 Exclusion Area Control - For any activity unrelated to facility operation conducted within the exclusion area, the applicant should identify the nature of his authority to determine all activities, including authority for the exclusion of personnel and property. Where the exclusion area is traversed by a highway, waterway, or railroad, the applicant should describe the arrangements made to control traffic in the event of an emergency.

2.1.2.2 Boundaries for Establishing Effluent Release Limits - The site description should clearly define the boundary line on which technical specification limits on the release of gaseous effluents will be

* As found on U.S. Geological Survey topographical maps.

based. This boundary line (which may or may not be the same as the plant property lines or the exclusion area boundary line) demarcates the area, access to which will be actively controlled for purposes of protection of individuals from exposure to radiation and radioactive materials. The degree of access control required is such that the licensee is able to fulfill his various obligations with respect to the requirements of 10 CFR Part 20, "Standards for Protection Against Radiation." 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 nearby rivers and lakes. Distances from plant effluent release points to the boundary line should be defined clearly.

2.1.3 Population and Population Distribution

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

2.1.3.1 Population Within Ten Miles - On a map of suitable scale which identifies places of significant population grouping, such as cities and towns within the 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 cardinal compass points (e.g., north, north-northeast, northeast, etc.). Within each area thus formed by the concentric circles and radial lines the current resident population should be specified, as well as the projected population by decade for at least four decades. Describe the basis for the projection.

2.1.3.2 Population Between 10 and 50 Miles - A map of suitable scale for these distances should be used in the same manner as described in 2.1.3.1 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 Low Population Zone - The low population zone (as defined in 10 CFR Part 100) and the basis for its selection should be specified. The population within the zone should be described in a manner similar to that described in 2.1.3.1 and 2.1.3.2, or presented in tabular form.

2.1.3.4 Transient Population - Variations in population on a seasonal basis should be described and, where appropriate, variations in population distribution during the working day should be discussed, particularly where significant shifts in population or population distribution may occur within the low population zone.

2.1.3.5 Population Center - The nearest population center (as defined in 10 CFR Part 100) should be specified and its population, direction, and distance from the reactor provided.

2.1.3.6 Public Facilities and Institutions - Any public facilities such as schools, hospitals, prisons, and parks within ten miles of the site should be identified and located with respect to the reactor, and their transient or permanent populations discussed.

2.1.4 Uses of Adjacent Lands and Waters

Land uses and uses of nearby bodies of water should be described in the application. Lands devoted to agricultural uses should be described in the context of principal food products, and acreage and yields. The nearest location suitable for dairying should be identified. The description of water uses should include extent of commercial and sport fishing, species and yields of fish taken and relative abundance, and commercial and recreational uses.

Sufficient data should be provided in this subsection regarding food crops and edible aquatic biota, in conjunction with estimated releases of radioactivity in gaseous and liquid effluents, to permit estimates to be made in Chapter 11 of the range of maximum potential annual radiation doses to individuals and to the population resulting from the principal radionuclides in discharged effluents.

2.2 Nearby Industrial, Transportation and Military Facilities

The purpose of this section is to establish whether the nuclear facility is designed to withstand safely the effects of potential accidents at, or as a result of the presence of, other industrial, transportation and military installations or operations in the vicinity* of the site which may have a potentially significant effect on the safe operation of the nuclear facility. These items should be re-evaluated at the time of the operating license review (FSAR), if any significant changes have occurred.

2.2.1 Locations and Routes

Provide a map showing all military bases, missile sites, manufacturing plants, chemical plants and storage facilities, airports, transportation routes (land and water), and oil and gas pipelines and tank farms. Include a description of military firing ranges and nearby airplane low level flight and landing patterns.

---

* All activities within five miles of the site should be considered. Activities at greater distances should be described and evaluated as appropriate to their significance.

2.2.2 Descriptions

A description of products manufactured, stored, or transported should be provided, as should the maximum quantities of hazardous material likely to be processed, stored, or transported.

2.2.3 Evaluations

Based on the information provided in subsections 2.2.1 and 2.2.2, a safety evaluation should be made for each of the activities including consideration of the following aspects as applicable.

For nuclear plants located on navigable waterways, the evaluation should consider the potential effects of impacts on the plant cooling water intake structures by the maximum size and weight of barges or ships that normally pass the site. (If the plant is located in a region in which low temperatures are experienced, discuss the protection provided to the intake structures against ice blockage and/or damage.) The effects of accidental upstream releases of corrosive liquids or oil on the intake structures should be evaluated.

The effects of explosion of chemicals, flammable gases, or munitions should be considered. If large natural gas pipelines cross, or pass close to the nuclear plant, explosions from this source should be evaluated. In situations where stone quarries are located near the site, consider the effect of detonation of the maximum amount of explosives that is permitted to be stored.

The potential effects of fires in adjacent oil and gasoline plants or storage facilities, adjacent industries, brush and forest fires and from transportation incidents should be evaluated. Evaluate the potential effects of accidental releases of toxic gases (e.g. chlorine) from onsite storage facilities, nearby industries and transportation accidents. The effect of expected airborne pollutants on critical reactor facility components should be evaluated to show the adequacy of the design, materials, construction, and operating procedures.

For sites in the vicinity of airports, evaluate the potential effects of aircraft impacts on the reactor facility, taking into account aircraft size, weight, and fuel loading.

In the event high natural-draft cooling towers or other tall structures such as discharge stacks are used on site, evaluate the potential for damage to equipment or structures important to reactor safety in the event of collapse.

2.3 Meteorology

This section should provide a meteorological description of the site and its surrounding areas, and sufficient data to describe the meteorological characteristics of the site. The information should be sufficient to permit an independent evaluation by the staff of the meteorological effects.

2.3.1 Regional Meteorology

2.3.1.1 Data Sources - Provide references to the climatic atlases and regional climatic summaries used.

2.3.1.2 General Climate - Describe the general climate of the region including the interplay between synoptic scale processes and terrain characteristics of the region.

2.3.1.3 Severe Weather - Provide the intensity and frequency of occurrence of heavy precipitation (rain and snow), hail, ice storms, thunderstorms, tornadoes, strong winds and high air pollutio potential.

2.3.2 Local Meteorology

2.3.2.1 Data Sources - Provide National Weather Service (NOAA) station summaries and other meteorological data which are indicative of site characteristics.

2.3.2.2 Normal and Extreme Values of Meteorological Parameters - Provide monthly summaries of wind (direction and speed combined), temperature, atmospheric water vapor (absolute and relative), precipitation (rain and snow), fog and atmospheric stability (if available).

2.3.2.3 Potential Influence of the Plant and Its Facilities on Local Meteorology - Discuss and provide an evaluation of the potential modification of the normal and extreme values of meteorological parameters described in 2.3.2.2 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).

2.3.2.4 Topographical Description - Provide a map showing the topographic features (as modified by the plant) within at least a five mile radius of the plant, and topographic cross sections in the 16 compass point sectors radiating from the plant. Include discussion of the effect of topography on short-term and long-term diffusion estimates from elevated release points, where appropriate.

2.3.3 Onsite Meteorological Measurements Programs

Provide a description of the preoperational and operational programs for meteorological measurements at the nuclear plant site, including measurements made, locations and elevations of measurements, description of instruments used, calibration and maintenance of instruments, data output and recording systems and data analysis procedures. (Additional guidance on acceptable onsite meteorological measurements programs is being developed in an AEC Safety Guide now in preparation.)

2.3.4 Short Term (Accident) Diffusion Estimates

2.3.4.1 Basis - Provide conservative estimates of atmospheric dilution factors at the site boundary and the outer boundary of the low population zone for appropriate time periods to 30 days after an accident, based on meteorological data.

2.3.4.2 Calculations - Describe the diffusion equations and the parameters used in the diffusion estimates.

2.3.5 Long Term (Routine) Diffusion Estimates

2.3.4.1 Basis - Provide realistic estimates of atmospheric dilution [illegible]

2.3.4.2 Calculations - Describe the diffusion equations and parameters used in the diffusion estimates.

2.4 Hydrology

The following subsections should contain sufficient information to allow an independent hydrologic engineering review to be made of all hydrologically related design bases, performance requirements, bases for design and operating procedures for structures, systems and components important to safety as a result of the following phenomena: (a) runoff type floods up to and including the probable maximum flood; (b) surges and wave action; (c) tsunamis; (d) artificial floods due to dam failures or landslides; (e) low water and/or drought effects on capability of cooling water supplies; (f) ice blockage of cooling water sources and ice jam flooding; (g) channel diversions of cooling water sources; (h) dilution and dispersion characteristics of the normal and accidental release hydrosphere relating existing and potential future users of surface and ground water resources.

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities - Describe the site and all safety-related elevations, structures, exterior accesses thereto and safety related equipment and systems from the standpoint of hydrologic considerations.

Provide a topographic map of the site and indicate thereon any proposed changes to natural drainage features.

2.4.1.2 Hydrosphere - Describe the location, size, shape and other hydrologic characteristics of streams, rivers, lakes, shore regions and groundwater environments influencing plant siting. Include a description of upstream and downstream river control structures, and provide a regional topographic map showing the major hydrologic features. List the owner, location, and rate of use of surface water users whose intakes could be adversely affected by accidental or normal releases of contaminants. Refer to subsection 2.4.13.2 for the tabulation of ground water users.

2.4.2 Floods

2.4.2.1 Flood History - Provide a synopsis of the flood history (date, level, peak discharge, etc.) 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 significant detrimental effects. Include river or stream floods, surges, tsunamis, dam failures, ice jams, etc.

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 nuclear power plants should be based on the highest calculated flood water level elevations and flood wave effects resulting from analysis of several different hypothetical floods. All possible flood conditions up to and including the highest and most critical flood level resulting from any of several different probable maximum events are to be considered as the basis for the design protection level for safety related components and structures of nuclear power plants. The probable maximum water level from a stream flood, surge, 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. Other possibilities are the flood level resulting from the most severe flood wave at the plant site caused by an upstream landslide, dam failure or dam breaching resulting from a seismic or foundation disturbance, or inadequate design capability. The effects of coincident wind generated wave action should be superimposed on the applicable flood level. The assumed hypothetical conditions are to be evaluated both statically and dynamically to determine the design flood protection level. The topical information required is generally outlined in subsections 2.4.3 through 2.4.6, but the type of events considered and the controlling event should be summarized in this subsection.

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

Describe the PMF defined by the Corps of Engineers as the "hypothetical flood characteristics (peak discharge, volume, and hydrograph shape) that

are considered to be the most severe "reasonably possible" at a particular location, based on relative comprehensive hydrometeorological analyses of critical runoff-producing precipitation (and snowmelt, if pertinent) and hydrologic factors favorable for maximum flood runoff." PMF determinations are usually prepared by estimating "probable maximum" precipitation (PMP) amounts over the subject drainage basin, in critical periods of time, and computing the residual runoff hydrograph likely to result with critical conditions of ground wetness and related factors. Estimates of the PMF are usually based on the observed and deduced characteristics of flood-producing storms and associated hydrologic factors, modified on the basis of hydrometeorological analyses to represent the most severe runoff conditions considered to be "reasonably possible" in the particular drainage basin under study. In addition to determining the PMF for adjacent large rivers or streams, a local PMF should be estimated for each local drainage course which can influence safety related facilities. 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 theoretically greatest 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 flood-producing storms in the general region of the drainage basin under study, and certain modifications and extrapolations of historical data to reflect more severe rainfall-runoff relations than actually recorded, insofar as these are deemed "reasonably possible" as occurrence 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 in the area such as reference to National Weather Service and Corps of Engineers determinations), time distributions, orographic effects, storm centering, seasonal effects, 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 studies in the region, or by presenting detailed storm-runoff studies.

2.4.3.3 Runoff Model - Describe the hydrologic response characteristics of the watershed to precipitation (such as unit hydrographs), verification from historic floods or synthetic procedures, the nonlinearity of the model due to high rainfall rates, and provide a description of sub-basin drainage areas (including a map), their sizes and topographic features of watersheds. Include a tabulation of all drainage areas, and runoff, reservoir and channel routing coefficients.

2.4.3.4 Probable Maximum Flood Flow - Present the PMF runoff hydrograph as defined as resulting from the probable maximum precipitation (and snow-melt, if pertinent) which considers the hydrologic characteristics of the potential influence of existing and proposed upstream dams and river structures for regulating or increasing the water level. If such dams are designed to withstand a PMF, their influence on the regulation of water flow and levels shall be considered; however, if a dam is not designed or constructed to withstand the PMF (or inflow from an upstream dam failure) the maximum water flows and resulting static and dynamic effects from the failure of the dam by breaching should be included in the PMF estimate. Discuss the PMF stream-course response model and its ability to compute floods of various magnitudes up to the severity of a PMF. Present any reservoir and channel routing assumptions with appropriate discussions of initial conditions, outlet works (both uncontrolled and controlled), spillway (both uncontrolled and controlled), the ability of any dams to withstand coincident reservoir wind wave action (including discussions of set-up, the significant wave height, the maximum wave height, and runup), the wave protection afforded, and the reservoir design capacity (i.e., the capacity for PMF and coincident wind wave action). Finally, provide the estimated PMF discharge hydrograph at the site and, when available, provide a similar hydrograph without upstream reservoir effects.

2.4.3.5 Water Level Determinations - Describe the translation of the estimated peak PMF discharge to elevation using cross section and profile data, reconstitution of historical floods (with consideration of high water marks and discharge estimates), standard step 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 the runup, wave heights, and resultant static and dynamic effects of wave action on each safety related facility from wind generated activity coincident with the peak PMF water level.

2.4.4 Potential Dam Failures (Seismically Induced)

Discuss and evaluate the effects of potential seismically induced dam failures on the upper limit of flood capability in streams and rivers. Consider the potential influence of upstream dams and river structures for regulating or increasing the water level. The maximum water flow and level resulting from failure of a dam by seismically induced breaching under the most severe probable modes of failure should be taken into account, including the potential for subsequent downstream domino-type failures due to flood waves. The simultaneous occurrence of the PMF and an earthquake capable of failing the upstream dams is not considered, since each of these

events considered singly has a low probability of occurrence. The suggested worst conditions at the dam site are to be evaluated by considering (1) a 25-year flood with full reservoirs coincident with an earthquake determined by a procedure similar to that used to determine the characteristics of the Safe Shutdown Earthquake* and (2) a standard-project flood or one-half the probable maximum flood (as defined by the Corps of Engineers) with full reservoirs coincident with the maximum earthquake determined on the basis of historic seismicity. Where downstream dams also regulate cooling water supplies, their potential failures also should be considered.

2.4.4.1 Reservoir Description - Include the locations of existing or proposed dams (both upstream and downstream) that influence conditions at the site, drainage areas above reservoirs, descriptions of types of structures, all appurtenances, ownership, seismic design criteria, and spillway design criteria.

2.4.4.2 Dam Failure Permutations - Discuss the locations of dams (both upstream and downstream), potential modes of failure and results of seismically induced and other types of dam failures that could cause the most critical conditions (floods or low water) with respect to the site for such an event. Consideration should be given to possible landslides, antecedent reservoir levels and river flows at the coincident flood peak (base flow). Present the determination of the peak flow rate at the site for the worst possible dam failure, and summarize an analysis to show that the presented condition is the worst permutation. Include the description of all coefficients and methods used.

2.4.4.3 Unsteady Flow Analysis of Potential Dam Failures - In determining the effect of dam failures at the site, the analytical methods presented should be applicable to artificial large floods with appropriately acceptable coefficients, and should also consider floodwaves through reservoirs downstream of failures. Domino-type failures due to flood waves should be considered where applicable. Discuss estimates of base flow and flood wave effects which are included to attenuate the dam failure flood wave downstream.

2.4.4.4 Water Level at Plant Site - Present the backwater or unsteady flow computation leading to the water elevation estimate for the most critical upstream dam failure, and discuss its reliability. Superimpose wind wave conditions that may occur simultaneously in a manner similar to that described in subsection 2.4.3.6.

---

* Refer to 10 CFR Part 100, proposed Appendix A.

2.4.5 Probable Maximum Surge Flooding

2.4.5.1 Probable Maximum Winds and Associated Meteorological Parameters - The mechanism is defined as a hypothetical hurricane or other cyclonic type wind storm that might result from the most severe combinations of meteorological parameters that are considered reasonably possible in the region involved, if the hurricane or other type wind storm should approach the point under study along a critical path and at optimum rate of movement. The determination of probable maximum meteorological winds 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" of occurrence on the basis of meteorological reasoning and should be presented in detail. The probable maximum conditions are the most severe combinations of hydrometeorological parameters (such as the meteorological characteristics of the probable maximum hurricane as reported by the U.S. National Oceanic and Atmospheric Administration in their unpublished report HUR 7-97*) considered reasonably possible that would produce the surge which has virtually no risk of being exceeded. This hypothetical event, as for other storm types, 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 assure that the parameters presented are the most reasonable severe combination.

2.4.5.2 Surge History - Discuss the proximity of the site to large bodies of water for which surge-type flooding can reach the site. The probable maximum water level (surges) for shore areas adjacent to large water bodies is the peak of the hypothetical surge stage hydrograph (still water levels), and coincident wave effects based on relative comprehensive hydrometeorological analyses resulting from the probable maximum meteorological criteria (such as hurricanes or other cyclonic wind storms) in conjunction with the critical hydrological characteristics that produce the probable maximum water level at a specific location. The effects of the probable maximum storms are superimposed on the coincidental maximum annual astronomical ambient tide levels, and associated wave action, to determine the effects of water level and wave action on structures. Provide a description of the surge history in the site region.

---

*This report, HUR-7-97, "Interim Report - Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States," is available upon request from the Hydrometeorological Branch, Office of Hydrology, NOAA, 8060 13th Street, Silver Spring, Md., 20910.

2.4.5.3 Surge Sources - Discuss considerations of hurricanes, frontal (cyclonic) type wind storms, moving squall lines, and surge mechanisms which are possible and applicable to the site. Include the antecedent water level (with reference to the spring tide for coastal locations, the average monthly high water for lakes, and a forerunner where applicable), the determination of the controlling storm surge (include the probable maximum meteorological parameters such as the storm track, wind fields, the fetch or direction of approach, bottom effects, and verification with historic events), the method used and results of the computation of the probable maximum surge hydrograph (graphical presentation).

2.4.5.4 Wave Action - Discuss the wind generated activity which can occur coincidentally with a surge, or independently thereof. Estimates of the wave period, the significant wave height and elevations, the maximum wave height and elevations, with the coincident water surge hydrograph should be presented. Specific data should be presented on the largest breaking wave height, setup, and runup that can reach each safety-related facility.

2.4.5.5 Resonance - Discuss the possibility of oscillations of waves at natural periodicity, such as lake reflection and harbor resonance phenomena, and any affects at the site.

2.4.5.6 Runup - Provide estimates of wave runup on plant facilities. Include a discussion of the water levels on each affected facility and the protection to be provided against static effects, dynamic effects, and splash. Cross reference 2.4.5.4 above for breaking waves.

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

2.4.6 Probable Maximum Tsunami Flooding

For sites adjacent to coastal areas, discuss historical tsunamis, either recorded or translated and inferred which provide information for use in determining the probable maximum water levels, and the geoseismic generating mechanisms available with appropriate references to section 2.5. The underwater geoseismic activity causes high speed, long period waves (tsunamis) that may produce catastrophic coastal damage even after being propagated thousands of miles. By far, the areas most susceptible to tsunamis are those bordering the Pacific, although their possible occurrence along the Gulf of Mexico and South Atlantic Coastlines should be recognized.

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 (such as fractures, faults, land slide potential, volcanism, etc.) in determining the limiting tsunami producing mechanism. The geoseismic investigations required 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 investigated that could produce the controlling maximum tsunami at the site from both local or distant generating mechanisms. Such considerations as the orientation of the site relative to the earthquake epicenter or generating mechanism, shape of the coastline, off-shore land areas, hydrography, stability of the coastal area (proneness of sliding), etc., should be factored into the analysis.

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

2.4.6.3 Source Tsunami Wave Height - Provide estimates of the maximum tsunami wave height possible at each major local generating source considered and the maximum offshore deepwater tsunami height from distant generators. Discuss the controlling generators for both locally and distantly generated tsunamis.

2.4.6.4 Tsunami Height Offshore - Provide estimates of the tsunami height in deep water adjacent to the site, or before bottom effects appreciably alter wave configuration, for each major generator.

2.4.6.5 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 of the analysis used to translate tsunami waves from offshore generator locations, or in deep water, to the site, and antecedent conditions. Provide, where possible, verification of the techniques and coefficients used by reconstituting tsunamis of record.

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

2.4.7 Ice Flooding

Present design criteria for protection of safety-related facilities from the most severe ice jam flood, wind-driven ice ridges, or ice-produced forces that are reasonably possible and could affect safety-related plant facilities with respect to adjacent rivers, streams, lakes, etc., and the location and proximity of such facilities to ice generating mechanisms. Describe the regional ice and ice jam formation history.

2.4.8 Cooling Water Canals and Reservoirs

2.4.8.1 Canals - Present the design bases for capacity and protection of canals against wind waves with acceptable freeboard, and (where applicable) the ability to withstand a probable maximum flood, surge, etc.

2.4.8.2 Reservoirs - Provide the design bases for capacity (reference subsection 2.4.11), the PMF design capability including wind wave protection, emergency storage evacuation (low level outlet and emergency spillway), with verified runoff models (unit hydrographs), flood routing, emergency spillway design, and outlet protection.

2.4.9 Channel Diversions

Discuss the potential for the upstream diversion or rerouting of the source of cooling water, such as river cutoffs, ice jams, or subsidence, with respect to historical and topographical evidence in the region. Present the history of flow diversions in the region. Describe available alternative cooling water sources in the event 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, or reference appropriate discussions in other sections of the SAR, required to assure that safety-related facilities will be capable of surviving all possible flood conditions up to and including the controlling severe event at the site.

2.4.11 Low Water Considerations

2.4.11.1 Low Flow in Rivers and Streams - Estimate the probable minimum flow level resulting from the most severe drought considered reasonably possible in the region as such conditions may affect the

source of cooling water and/or the ability of water related ultimate heat sinks to perform adequately under severe hydrometeorological conditions.

2.4.11.2 Low Water Resulting from Surges - Determine the surge or tsunami caused low water level that could occur from probable maximum meteorological or geoseismic conditions. Include a description of the probable maximum wind storm, its track, associated parameters, antecedent conditions (see 2.4.5.4 above), and the computed low water level, or tsunami conditions applicable. Also consider, where applicable, ice formation, or ice jams causing low flow, as such conditions may affect the cooling water source.

2.4.11.3 Historical Low Water - Discuss historical low water controls, minimum stream flows or minimum surges and elevations, and probabilities (unadjusted for historical controls and adjusted for 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 Control - Provide the estimated flow rate, durations and levels for probable minimum flow conditions considering future uses.

2.4.11.5 Plant Requirements - Present the required minimum cooling water flow, the sump invert elevation and configuration, the minimum design operating level, pump submergence elevations (operating heads), effluent submergence and mixing and dispersion design bases. Discuss the capability of cooling water pumps to supply sufficient water during periods of extreme low water level.

2.4.11.6 Dependability Requirements - Describe the ability to provide warning of impending low flow to allow switching to alternative sources where applicable. Compare minimum flow and/or level estimates with plant requirements and describe any available low water safety factor.

2.4.12 Environmental Acceptance of Effluents

Describe the ability of the environment to disperse and dilute normal and inadvertent or accidental releases of radioactive effluents for the full range of anticipated operating conditions. Present the applicable design bases for effluent facilities to meet design requirements. Refer to subsections 2.4.1.2 and 2.4.13.2 for the locations and users, respectively, of surface and ground waters.

2.4.13 Groundwater

2.4.13.1 Description and On-site Use - Describe the regional and local groundwater aquifers, formations, sources, and sinks. Describe the type of ground water use, well, pump and storage facilities, and flow requirements of the plant by type of use.

2.4.13.2 Sources - Describe present regional use, and projected future use; tabulate existing users (amounts, location and drawdown) and piezometric levels; indicate flow directions and gradients; and discuss the reversibility of ground water flow and the effects of potential future use on the flow rates, gradients and groundwater levels beneath the site. Note any potential ground water recharge area within influence of the plant.

2.4.13.3 Accident Effects - Provide an evaluation of the dispersion and dilution capability of the groundwater environment with respect to existing users and future users under operating and accident conditions.

2.4.13.4 Monitoring or Safeguard Requirements - Discuss the need for procedures and safeguards to protect groundwater users if the potential for contamination is high. Present preliminary plans for such safeguards and/or monitoring.

2.4.14 Technical Specification and Emergency Operation Requirements

Describe any emergency protective measures designed to minimize the water associated impact of adverse hydrologically related events on safety related facilities. Describe the manner in which these requirements will be incorporated into appropriate Technical Specifications and/or Emergency Procedures. Discuss the need for any Technical Specifications for plant shutdown to minimize the consequences of an accident due to hydrologically associated phenomena. In the event emergency procedures are to be utilized to meet safety requirements due to hydrologically related events, present appropriate water levels, lead times available and indicate what type of action would be taken.

2.5 Geology and Seismology

This section should provide information regarding the seismic and geologic characteristics of the site. Guidance is provided in proposed Appendix A to 10 CFR Part 100, "Seismic and Geologic Siting Criteria for Nuclear Power Plants" (published for comment in the Federal Register, Vol. 36,

No. 228, November 25, 1971), which sets forth the principal seismic and geologic considerations that guide the regulatory staff in its evaluation of the acceptability of sites and seismic design bases.

2.5.1 Basic Geologic and Seismic Data

The following basic data should be included concerning the geology and seismology of the site and the region surrounding the site.

(1) Description of the physiography of the region and of the site and maps showing the physiographic features of the site and the surrounding region. The maps should include the site location.

(2) Geologic and tectonic maps of the region surrounding the site.

(3) Geologic map of the site area which shows surface geology and which includes the locations of major structures of the nuclear power plants, including all Category I structures.

(4) Structural geologic map of the site area which shows bedrock contours and which includes the location of major structures of the nuclear power plant, including all Category I structures.

(5) Geologic profiles showing the relationship of the major foundations of the nuclear power plant to subsurface materials, including ground water, and the significant engineering characteristics of the subsurface materials.

(6) History of ground water fluctuations beneath the site and a discussion of ground water conditions during construction of the nuclear power plant and during plant life.

(7) A plot plan showing the locations of all Category I structures of the nuclear power plant and the locations of all borings, trenches, and excavations along with a description, logs, and maps of the borings, trenches, and excavations, as necessary to indicate the results.

(8) Results of seismic refraction and reflection surveys, and shear wave velocity and uphole velocity surveys.

(9) Summary of static and dynamic soil and rock properties at the site including grain-size classification, Atterberg limits, water content, density, shear strength, relative density, shear modulus, Poisson's ratio, bulk modulus, damping.

(10) Plan and profile drawings showing the extent of excavations and backfill planned at the site and compaction criteria for all engineered backfill.

(11) The detailed safety related criteria and the computed factors of safety for the materials underlying the foundations for all Category I nuclear power plant structures and for all Category I embankments under dynamic conditions combined with adverse hydrologic conditions.

2.5.2 Vibratory Ground Motion

Information should be presented to describe how the design basis for vibratory ground motion (Safe Shutdown Earthquake) was determined. The following specific information should also be included:

(1) Describe the lithologic, stratigraphic, and structural geologic conditions of the site and the region surrounding the site, including its geologic history.

(2) Identify tectonic structures underlying the site and the region surrounding the site.

(3) Describe physical evidence concerning the behavior during prior earthquakes of the surficial geologic materials and the substrata underlying the site from the lithologic, stratigraphic, and structural geologic studies.

(4) Describe the static and dynamic engineering properties of the materials underlying the site. Included should be properties needed to determine the behavior of the underlying material during earthquakes and the characteristics of the underlying material in transmitting earthquake-induced motions to the foundations of the plant, such as seismic wave velocities, density, water content, porosity, and strength.

(5) List all historically reported earthquakes which have affected, or which could be reasonably expected to have affected the site, including the date of occurrence and the following measured or estimated data: magnitude or highest intensity, and a plot of the epicenter or region of highest intensity. Where historically reported earthquakes could have caused a maximum ground acceleration of at least one-tenth the acceleration of gravity (0.1g) at the foundations of the proposed nuclear power plant structures, the acceleration or intensity and duration of ground shaking at these foundations should also be estimated. Since earthquakes have been reported in terms of various parameters, such as magnitude, intensity

at a given location, and effect on ground, structures, and people at a specific location, some of these data may have to be estimated by use of appropriate empirical relationships. Where appropriate, the comparative characteristics of the material underlying the epicentral location or region of highest intensity and of the material underlying the site in transmitting earthquake vibratory motion should be considered.

(6) Provide a correlation of epicenters or regions of highest intensity of historically reported earthquakes, where possible, with tectonic structures, any part of which is located within 200 miles of the site. Epicenters or regions of highest intensity which cannot be reasonably correlated with tectonic structures should be identified with tectonic provinces, any part of which is located within 200 miles of the site.

(7) For faults, any part of which is within 200 miles of the site and which may be of significance in establishing the Safe Shutdown Earthquake, determine whether these faults should be considered as active faults.

(8) For faults, any part of which is within 200 miles of the site which may be of significance in establishing the Safe Shutdown Earthquake and which are considered as active faults, determine: the length of the fault; the relationship of the fault to regional tectonic structures; and the nature, amount, and geologic history of displacements along the fault, including particularly the estimated amount of the maximum Quaternary displacement related to any one earthquake along the fault.

(9) The historic earthquakes of greatest magnitude or intensity which have been correlated with tectonic structures should be determined. For active faults, the earthquake of greatest magnitude related to the faults should be determined taking into account geologic evidence. The vibratory ground motion at the site should be determined assuming the epicenter of the earthquakes are situated at the point on the structures closest to the site.

(10) Where epicenters or regions of highest intensity of historically reported earthquakes cannot be related to tectonic structures but are identified with tectonic provinces in which the site is located, the accelerations at the site should be determined assuming that these earthquakes occur adjacent to the site.

(11) Where epicenters or regions of highest intensity of historically reported earthquakes cannot be related to tectonic structures but are identified with tectonic provinces in which the site is not located, the

accelerations at the site should be determined assuming that the epicenters or regions of highest intensity of these earthquakes are located at the closest point to the site on the boundary of the tectonic province.

(12) The earthquake producing the maximum vibratory accelerations at the site should be designated the Safe Shutdown Earthquake for vibratory ground motion. The Safe Shutdown Earthquake should be defined by response spectra corresponding to the maximum vibratory accelerations.

(13) The Operating Basis Earthquake, where one is selected by the applicant, should be defined by response spectra.

2.5.3 Surface Faulting

Information should be presented which describes whether and to what extent the nuclear power plant need be designed for surface faulting. The following specific information should also be included:

(1) Describe the lithologic, stratigraphic, and structural geologic conditions of the site and the area surrounding the site, including its geologic history (or cross-reference subsection 2.5.2).

(2) Determine the geologic evidence of fault offset at or near the ground surface at or near the site.

(3) For faults greater than 1,000 feet long, any part of which is within 5 miles of the site, determine whether these faults should be considered as active faults.

(4) List all historically reported earthquakes which can be reasonably associated with active faults greater than 1,000 feet long, any part of which is within 5 miles of the site, including the date of occurrence and the following measured or estimated data: Magnitude or highest intensity, and a plot of the epicenter or region of highest intensity.

(5) Provide a correlation of epicenters or regions of highest intensity of historically reported earthquakes with active faults greater than 1,000 feet long, any part of which is located within 5 miles of the site.

(6) For active faults greater than 1,000 feet long, any part of which is within 5 miles of the site, determine: the length of the fault; the relationship of the fault to regional tectonic structures; the nature, amount, and geologic history of displacements along the fault; and the outer limits of the fault established by mapping fault traces for 10 miles along its trend in both directions from the point of its nearest approach to the site.

(7) Determine the zone requiring detailed faulting investigation.

(8) Where the site is located within a zone requiring detailed faulting investigation, the results of this investigation, to determine the need to take into account surface faulting in the design of the nuclear power plant, should be presented.

(9) Where it is determined that surface faulting need not be taken into account, sufficient data should be presented to justify the determination clearly.

(10) Where it is determined that surface faulting need be taken into account, the design basis for surface faulting should be presented.

2.5.4 Stability of Subsurface Materials

Information should be presented concerning the stability of soils and rock underneath the nuclear power plant foundations during the vibratory motion associated with the Safe Shutdown Earthquake. Evaluation of the following geologic features which could affect the foundations should be presented:

(1) Areas of actual or potential surface or subsurface subsidence, uplift, or collapse resulting from:

(i) Natural features such as tectonic depressions and cavernous or karst terrains, particularly those underlain by calcareous or other soluble deposits;

(ii) Man's activities, such as withdrawal or addition of subsurface fluids, or mineral extraction;

(iii) Regional warping.

(2) Deformational zones, such as shears, joints, fractures and folds, or combinations of these features.

(3) Zones of alteration or irregular weathering profiles, and zones of structural weakness composed of crushed or disturbed materials.

(4) Unrelieved residual stresses in bedrock.

(5) Rocks or soils that might be unstable because of their mineralogy, lack of consolidation, water content, or potentially undesirable response to seismic or other events. (Seismic response characteristics to be considered include liquefaction, thixotrophy, differential consolidation, cratering, and fissuring.)

2.5.5 Slope Stability

Information and appropriate substantiation should be presented concerning the stability of all slopes, both natural and artificial, the failure of which could adversely affect the nuclear power plant, during the occurrence of the Safe Shutdown Earthquake.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

This chapter of the Safety Analysis Report should identify, describe and discuss the principal architectural and engineering design criteria that represent the broad frame of reference within which the more detailed design effort of those structures, components, equipment, and systems important to safety is to proceed and against which attainment of the design objective will be judged.

Where the need arises in other chapters of the SAR to refer to design criteria included in this section, only cross referencing is necessary.

3.1 Conformance with AEC General Design Criteria

This section should discuss briefly the extent to which the design criteria for the facility structures, systems and components important to safety meet the AEC "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 or bases meet the criterion. In the discussion of each criterion, the sections of the SAR where more detailed information is presented to demonstrate compliance with the criterion should be referenced.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

Structures, systems, and components are classified for seismic design purposes as either Category I or Category II. Those structures, systems and components important to safety that are designed to remain functional in the event of a Safe Shutdown Earthquake (see Section 2.5) are designated as Category I. These structures, systems, and components are those necessary to assure:

(1) The integrity of the reactor coolant pressure boundary,

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

(3) The capability to prevent or mitigate the consequences of accidents which could result in potential off site exposures comparable to the guideline exposures of 10 CFR Part 100.

Those structures, systems, and components that are designed to remain operable in the event of the Operating Basis Earthquake, if it is proposed to continue to generate power, are designated as Category II.

This subsection of the SAR should provide a list of all Category I structures, components and systems to permit a determination to be made as to the general suitability of the classification given and the design approach being applied in the design of these structures.

Structures and systems which are partially Category I and partially in a lesser category should be listed and where necessary for clarity, the boundaries of the Category I portions should be shown on appropriate drawings.

For boiling water reactors, if the list of structures, components, and systems that have been designated as Category I does not include that portion of the main steam system extending from the outermost containment isolation valve up to the turbine casing and connected piping inclusive of the first valve (which is either normally closed or capable of automatic closure), submit justification for the proposed classification.

3.2.2 System Quality Group Classification

Provide a tabulation of and delineate on the Piping and Instrumentation Diagrams the system quality group classifications (see Table 3.2.2-1) for each pressure-containing component of (a) those applicable fluid systems relied upon to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, or to permit shutdown of the reactor and maintenance in the safe shutdown condition, and (b) other associated safety related systems.

3.3 Wind and Tornado Design Criteria

3.3.1 Wind Criteria

Provide the design wind velocity, recurrence interval, data sources and history, as well as the methods used in applying these wind loads to Category I structures as forces, and the techniques used in designing these structures for the wind loads. If the criteria are not applied to all Category I structures, justify any exclusions.

TABLE 3.2.2-1 Summary of Codes and Standards for Components of Water-Cooled Nuclear Power Units

Component	Code Classifications			
	Group A	Group B	Group C	Group D
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class A	ASME Boiler and Pressure Vessel Code, Section III, Class C	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 or Equivalent
0-15 Psig Storage Tanks	-	API-620 with the NDT Examination Requirements in Table NST-1, Class 2	API-620 with the NDT Examination Requirements in Table NST-1, Class 3	API-620 or Equivalent
Atmospheric Storage Tanks	-	Applicable Storage Tank Codes such as API-650, AWWAD100 or ANSI B 96.1 with the NDT Examination Requirements in Table NST-1, Class 2	Applicable Storage Tank Codes such as API-650 AWWAD100 or ANSI B 96.1 with the NDT Examination Requirements in Table NST-1, Class 3	API-650, AWWAD100 or ANSI B 96.1 or Equivalent
Piping	ANSI B 31.7, Class 1	ANSI B 31.7, Class II	ANSI B 31.7, Class III	ANSI B 31.1.0 or Equivalent
Pumps and Valves	Draft ASME Code for Pumps and Valves Class I. See Footnote (a)	Draft ASME Code for Pumps and Valves Class II. See Footnote (a)	Draft ASME Code for Pumps and Valves Class III.	Valves - ANSI B 31.1.0 or Equivalent Pumps - Draft ASME Code for Pumps Valves Class III or Equivalent

FOOTNOTE:

(a) All pressure-retaining cast parts shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified in the applicable class in the code.

3.3.2 Tornado Criteria

Provide the design parameters, applicable to the design tornado, such as rotational and translational velocity, design pressure differential and associated time interval, and the tornado-generated missile impact load and state whether the imposed loads will be applied simultaneously in establishing the tornado design. If the tornado design parameters are different from those that have been accepted for recently licensed nuclear power plants (i.e., 300 mph rotational, 60 mph translational and a 3 psi pressure differential in 3 seconds, and inclusion of tornado-generated missile impact loads, all applied simultaneously), justify the tornado design values used, by showing that the design using these values will provide a level of conservatism equivalent to that considered acceptable for previously licensed nuclear power plants and consistent with the state-of-the-art knowledge of tornadoes. Also, provide information to show that those structures not designed for tornado loads (including Category I structures, if any) will not as a result of possible failure under such loads, affect the ability of other Category I structures or systems to perform their intended design functions.

Describe the methods used to convert the tornado loadings into forces on the structures, including the distributions across the structures (e.g., on large structures such as reactor buildings usually a uniform 300 mph wind is taken over the entire surface, while on small structures such as stacks or pump houses a peak 360 mph wind is taken over the structure). Discuss the validity of the methods. If factored loads are used, then the basis for selection of the load factor used for tornado loading should be furnished.

3.4 Water Level (Flood) Design Criteria

This section should discuss the design bases with respect to the structural capability of Category I structures to withstand the static and dynamic forces associated with the design flood level established for the site, discussed in Section 2.4 of the SAR (i.e., for that condition or combination of conditions, such as the maximum probable flood and coincident wind wave activity, hurricane, tsunami, seiche or other phenomena, which has been predicted to result in the maximum water level at the site).

3.5 Missile Protection Criteria

This section should describe the design bases with respect to internal and external missiles for which the plant is analyzed and protected against. The discussion should consider: (a) missiles that might be generated

within the plant as a result of failure of rotating or pressurized components or equipment, (b) tornado-generated missiles, (c) missiles that might result from activities particular to a given site location (such as airports, nearby industry, transportation, etc.).

For each missile considered, give the design parameters of origin, impact velocity or energy and orientation, density and other pertinent assumptions used in the analysis, along with the bases for each. State which structures are involved in the analysis of missile damage protection and give the bases for the selections made. The analytical techniques should be described, and the level of conservatism should be discussed.

3.6 Criteria for Protection Against Dynamic Effects Associated with a Loss-of-Coolant Accident

This section should describe the measures that have been used to assure that the containment vessel and all essential equipment within the containment, including components of the reactor coolant pressure boundary, engineered safety features, and equipment supports, have been adequately protected against the effects of blowdown jet forces, and pipe whip resulting from a loss-of-coolant accident. The description should include:

(a) Pipe restraint design requirements to prevent pipe whip impact.

(b) The features provided to shield vital equipment from pipe whip.

(c) The measures taken to physically separate piping and other components of redundant engineered safety features.

(d) A description of the analyses performed to determine that the failure of lines, with diameters of 3/4 inch or less, will not cause failure of the containment liner under the most adverse design basis accident conditions.

(e) The analytical methods which were used.

(f) Provide the design loading combinations, the design condition categories (normal, upset, emergency, and faulted) and design stress limits, applied to the supports and pipe whip restraints of all Category I components and piping of fluid systems. Identify the applicable design codes used. If the proposed design criteria allow plastic deformation of supports indicate whether this design approach includes the inelastic strain compatibility in the supports and supported components in a combined dynamic system analysis for all systems where the proposed stress and strain limits apply.

3.7 Seismic Design

3.7.1 Input Criteria

This subsection should discuss the input criteria for seismic design of the plant including the following specific information:

(1) Provide design response spectra (OBE and SSE) that account for earthquake duration and the effect of distances and depth between the seismic disturbances and the site. The design response spectra should also be based on amplification factors that are derived from existing earthquake records. Site seismic design response spectra which define the vibratory ground motions of the Safe Shutdown Earthquake at the elevations of the foundations of the nuclear power plant structures, as required in the Seismic and Geologic Siting Criteria (proposed Appendix A to 10 CFR Part 100), should be provided. In view of the limited data presently available on vibratory ground motions of strong earthquakes, the design response spectra should be developed from an envelope of spectra which are related to the vibratory motions caused by more than one earthquake, and should take into account the fact that representative response spectra obtained from these earthquake records show that for 2% damping peak amplification factors are in the range of 2.5 to 5.0 for the period range of 0.15 to 0.5 seconds, and that amplification factors are greater than 1.0 in the period range 0.03 to 0.15 seconds.

(2) The response spectra derived from the actual or synthetic earthquake time motion records used for design should be provided and should envelope the site seismic design response spectra appropriate for the nuclear power plant discussed in item (1) above. Provide a comparison, for all the damping values that are used in the design, of the response spectra derived from the time history and the site seismic design response spectra. The system period intervals at which the spectra values were calculated should be identified and criteria should be provided to demonstrate that these intervals are small enough to produce sufficiently accurate response spectra.

(3) The specific percentage of critical damping values used for all Category I and Category II structures, systems and components should be provided. The information should also include the type of construction or fabrication (e.g., prestressed concrete, welded pipe, etc.) and the applicable allowable design stress levels for these plant features.

(4) If a site dependent analysis is used to develop the shape of the site seismic design response spectra from bedrock time history or response spectra input, then the bases for this analytical approach should be provided. Specifically, the bases for use of in-situ soil measurements, soil layer location and bedrock earthquake records should

be provided. If the analytical approach used to determine the shape of the site seismic design response spectra neglects vertical amplification and possible slanted soil layers, the validity of these assumptions should be discussed. The influence of possible predominate thin soil layers on the analytical results should also be discussed.

(5) Provide a list of all soil-supported Category I and Category II structures, and identify the depth of soil over bedrock for each structure listed.

(6) If a simplified lumped mass and soil spring approach is used in the PSAR to characterize soil structure interaction, justification should be submitted for soil sensitive sites. The use of equivalent soil springs for the seismic-system mathematical models may produce a pronounced filtering of the ground motion response amplitudes and response frequencies due to sensitive soil parameters. Provide the basis for the use of a lumped parameter mathematical model with equivalent soil springs in lieu of a finite element model (or equivalent method), including the use of parametric studies which evaluate possible variations in the in-situ soil properties (e.g., moduli, density, and stress level).

3.7.2 Seismic System Analysis

This subsection should discuss the seismic system analyses performed for Category I and Category II structures and systems. The following specific information should be included:

(1) For all Category I and Category II structures, systems, and components (listed in Section 3.2.1), identify the methods of seismic analysis (modal analysis response spectra, modal analysis time history, equivalent static load, etc.) used for each of the items including the reactor core support structure. Include applicable stress or deformation criteria and descriptions (sketches) of the mathematical models used. If empirical methods (tests) are used in lieu of analysis, also provide the criteria and acceptance basis used to confirm the integrity of the structures, systems, components and equipment. Describe all seismic methods of analyses used.

(2) Provide the criteria used to lump masses for the seismic system analyses (system mass and compliance to component or bay characteristics and floor mass and compliance to equipment characteristics). Provide the procedures or criteria used to assure that all the required inputs and/or responses required by different design organizations for all Category I structures, systems, components and equipment are derived from either the seismic-system (multi-mass time history) method or equivalent theoretical or empirical analyses.

(3) The validity of a fixed base assumption in the mathematical models for the dynamic system analyses should be confirmed by providing summary analytical results that indicate that the rocking and translational response are insignificant. Include a brief description of the method, mathematical model and damping values (rocking vertical, translation and torsion) that have been used to consider the soil-structure interaction.

(4) Provide the methods and procedures used to couple the soil and the seismic-system structures and components in the event a finite element analysis is used in lieu of a lumped mass system model with soil springs.

(5) Indicate whether the modal response spectra multi-mass method of analysis is used to develop floor response spectra. Since component and floor input response spectra for various locations within the building structures and for major components are not directly obtainable by this method, evidence of its conservatism should be presented, either by demonstrating equivalency to a multi-mass time history method or by submitting other theoretical or experimental justification. Provide the stress and deformation basis for consideration of the differential seismic movement of interconnected components between floors.

(6) Describe the measures taken to consider the effects on floor response spectra (e.g., peak width and period coordinates) of expected variations of structural properties, dampings, soil properties, and soil-structure interactions.

(7) Justify the use of constant vertical load factors as vertical response loads for the seismic design of all Category I structures, systems, and components rather than a multi-mass dynamic analysis procedure, taking into account the following considerations:

(a) The possible combined horizontal and vertical amplified response loading for the seismic design of the building and floors.

(b) The possible combined horizontal and vertical amplified response loading for the seismic design of equipment and components, including the effect of the seismic response of the building and floors.

(c) The possible combined horizontal and vertical amplified response loading for the seismic design of piping and instrumentation, including the effect of the seismic response of the building, floors, supports, equipment, component, etc.

(8) Describe the method employed to consider the torsional modes of vibration in the seismic analysis of the Category I building structures. If static factors are used to account for torsional accelerations in the seismic design of Category I structures, justify this procedure in lieu of a combined vertical, horizontal, and torsional multi-mass system dynamic analysis.

(9) The use of both the modal analysis response spectrum and time history methods provides a check on the response at selected points in the station structure. Submit the responses obtained from both of these methods at selected points in the Category I structure to provide the basis for checking the seismic system analysis.

(10) Describe the analytical methods and procedures used for the seismic system analysis of dams that impound bodies of water to serve as heat sinks.

(11) Describe the design control measures instituted to assure that adequate seismic input (including any necessary feedback from structural and system dynamic analyses) is specified to vendors of purchased Category I components and equipment. Identify the responsible design groups or organizations who assure the adequacy and validity of the analyses and tests employed by vendors of Category I components and equipment, and describe the review procedures utilized by each group or organization.

(12) Provide the dynamic methods and procedures used to determine Category I structure overturning moments. Include a description of the procedures used to account for soil reactions and vertical earthquake effects.

(13) Provide the basis for simplified seismic analysis methods and procedures used for seismic designs, including the criteria used to avoid the predominate input frequencies produced by the response of buildings, supports, and components to the earthquake input.

(14) Provide the analysis procedure followed to account for the damping in different elements of the model of a coupled system. Include the criteria used to account for composite damping in a coupled system with different structural elements.

(15) Provide the criteria used to account for modal period variation in the mathematical models for Category I structures due to variations in material properties. Indicate the percentage increase in the resultant seismic loads.

(16) Provide the damping factors used for the seismic design of all Category I structures, systems, components, and equipment.

3.7.3 Seismic Subsystem Analysis

The discussion on the seismic subsystem analysis should include the following specific information:

(1) Describe the procedures used to account for the number of earthquake cycles during one seismic event, and specify the number of loading cycles for which Category I systems, components, and equipment are designed, including the expected duration of the seismic motions or the number of major motion peaks.

(2) Provide the basis for the selection of frequencies to preclude resonance by demonstrating that the earthquakes specified for the site, and building and component response characteristics either filter or preclude higher frequencies than the frequencies specified.

(3) If the term "root-mean-square basis" is used in describing the combination of modal responses, confirm that the responses are combined using the square root of the sum of the squares.

(4) Provide the criteria for combining modal responses (shears, moments, stresses, deflections, and/or accelerations) when modal frequencies are closely spaced and a response spectrum modal analysis method is used.

(5) If static loads equivalent to the peak of the floor spectrum curve are used for the seismic design of components and equipment, justify the use of peak spectrum values by demonstrating that the contribution of all significant dynamic modes of response under seismic excitation has been included in the analyses to be performed.

(6) Provide the design criteria and analytical procedures applicable to piping that take into account the relative displacements between piping support points, i.e., floors and components, at different elevations within a building and between buildings.

(7) Submit the basis for the methods used to determine the possible combined horizontal and vertical amplified response loading for the seismic design of piping and instrumentation, including the effect of the seismic response of the supports, equipment, and components.

(8) If a simplified dynamic analysis is used for Category I piping, indicate the magnitude by which the resonant periods of a selected piping span are removed from the predominate supporting building and component

periods. Submit a summary of typical results from the simplified dynamic methods and the dynamic response spectra analytical methods.

(9) Provide the criteria employed to account for the torsional effects of valves and other eccentric masses (e.g., valve operators) in the seismic piping analyses.

(10) With respect to Category I piping buried or otherwise located outside of the containment structure, describe the seismic design criteria employed to assure that allowable piping and structural stresses are not exceeded due to differential movement at support points, at containment penetrations, and at entry points into other structures.

(11) Describe the evaluation performed to determine seismic induced effects of Category II piping systems on Category I piping.

(12) Provide the criteria employed to determine the field location of seismic supports and restraints for Category 1 piping, piping system components, and equipment, including placement of snubbers and dampers. Describe the procedures followed to assure that the field location and characteristics of these supports and restraining devices are consistent with the assumptions made in the dynamic analyses of the system.

(13) Indicate the provisions taken to assure that any cranes located in the reactor building will not be dislodged from their rails in the event of seismic exitation.

3.7.4 Criteria for Seismic Instrumentation Program

With respect to the criteria for seismic instrumentation, the following should be provided:

(1) Discuss the seismic instrumentation provided and compare the proposed seismic instrumentation program with that described in AEC Safety Guide 12, "Instrumentation for Earthquakes." Submit the basis and justification for elements of the proposed program that differ substantially from Safety Guide 12.

(2) Provide a description of the seismic instrumentation such as peak recording accelerographs and peak deflection recorders, that will be installed in selected Category I structures and on selected Category I components. Include the basis for selection of these structures and components, the basis for location of the instrumentation, and the extent to which this instrumentation will be employed to verify the seismic analyses following a seismic event.

(3) Describe the provisions that will be employed to provide the value of the peak acceleration level experienced in the basement of the reactor containment structure to the control room operator within a few minutes after the earthquake. Include the basis for establishing the predetermined values for activating the readout of the accelerograph to the control room operator.

(4) Provide the criteria and procedures that will be used to compare measured responses of Category I structures in the event of an earthquake with the results of the system dynamic analyses. Include consideration of different underlying soil conditions or unique structural dynamic characteristics that may produce different dynamic responses of Category I structures at the site.

3.7.5 Seismic Design Control Measures

This section should describe the design control measures instituted to assure that adequate seismic input data (including any necessary feedback from structural and system dynamic analyses) are specified to vendors of purchased Category I components and equipment. Identify the responsible design groups or organizations that assure the adequacy and validity of the analyses and tests employed by vendors of Category I components and equipment. Provide a description of the review procedures utilized by each group or organization.

3.8 Design of Category I and Category II Structures

3.8.1 Structures Other than Containment

This section should discuss the design bases, criteria, and analytical techniques upon which the design of Category I and Category II structures, other than the containment structure, is based. Lengthy, detailed descriptions of specific topics not readily incorporated in the main text, such as detailed program descriptions, may be provided as appendices to the SAR. The following specific information should be provided:

(1) A physical description of each structure (listed in Section 3.2.1) should be furnished. The influence of any lesser category structure or component on the Category I structure should be described.

The design bases should be stated for each structure.

(2) Codes, specifications, regulations, safety guides, or other similar documents used in establishing or implementing design bases and methods, analytical techniques, material properties and quality control provisions should be listed. Any modifications, deletions or additions to these documents should be described.

(3) The load combinations used in analysis and design should be listed. If an analytical or design approach using load factors other than 1.0 is utilized, these load factors should be included, and reference should be made to the applicable section of the SAR that describes the approach used.

(4) The analytical techniques employed should be described. The descriptions should cover the general analysis for the loads and load combinations listed in item (3) above. Any techniques utilized which are not fully described by references given in item (2) above should be explained, and bases for use of the techniques furnished. The analyses for seismic and tornado loads should be explained in sufficient detail to permit understanding of the approaches taken, and the degree of conservatism available in the designs.

(5) An explanation of the design methods, calculated stresses and strains, and allowable stresses and strains should be furnished for the principal structural components. If deformations are permitted by design, then limits should be described which assure continued functional capability of the structure or any other Category I structure or component which interacts with the designed structure.

(6) All principal construction materials should be identified and described. Any material not readily identified by standard industry specifications should have its physical and mechanical properties described. Quality control procedures used during fabrication or installation should be furnished. Any construction procedures involving unusual techniques, or quality control standards in excess of normal construction practices should be outlined.

(7) Any structural preoperational testing procedures, other than those described in item (6) should be furnished. Any structural post-operation surveillance programs should also be furnished.

3.8.2 Containment Structure

This section should discuss the design bases, criteria, and analytical techniques upon which the design of the containment structure is based, including the following specific information:

(1) Present a physical description of the containment structure.

(2) The design bases for the containment structure should be provided, including the functional criteria for operation, accident containment, testing and surveillance. The design requirements with respect to external pressure loading should be described. In this regard discuss the utilization of vacuum breakers, or purge valves.

(3) Codes, specifications, regulations, safety guides, or other similar documents used in establishing or implementing design bases and methods, analytical techniques, material properties and quality control provisions should be listed. Any modifications, deletions or additions to these documents should be described.

(4) The load combinations used in analysis and design should be listed. If an analytical or design approach using load factors other than 1.0 is utilized, these load factors should be included, and reference made to the applicable section which describes the approach used.

(5) The analytical techniques employed should be described. The descriptions should cover the general analysis for the loads and load combinations listed in item (4) above. Any techniques utilized which are not fully described by references given in item (3) above should be explained, and bases for use of the techniques furnished. The analyses for seismic and tornado loads should be explained in sufficient detail to permit understanding of the approaches taken, and the degree of conservatism available in the designs.

(6) An explanation of the design methods, calculated stresses and strains, and allowable stresses and strains should be furnished for the principal structural components. If deformations are permitted by design, then limits should be described which assure continued functional capability of the containment structure or any other Category I structure or component which may interact with it. Provide a discussion of the design methods used for containment subcompartments enclosing such components as the reactor vessel (reactor cavity), the pressurizer, and steam generators. For those containment systems where vital subcompartments cannot be readily pressure tested, assurance should be provided that the structural design analysis of the subcompartments will be performed by two independent organizations or two independent and separate groups within the applicant's organization.

(7) All principal construction materials should be identified and described. Any material not readily identified by standard industry specifications should have its physical and mechanical properties described. Quality control procedures used during fabrication or installation should be furnished. Any construction procedures involving unused techniques, or quality control standards in excess of normal construction practices should be outlined.

(8) Structural preoperational testing procedures, other than those described in item (7) above, should be furnished. Any structural post-operation surveillance programs should also be furnished.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

This subsection should provide, as a minimum, the following specific information:

(1) Describe the vibration operational test program required by NB-3622.3, NC-3622, and ND-3611 of ASME Section III used to verify that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips, etc. Provide a list of the transient conditions and the associated actions (pump trips, valve actuations, etc.) that will be used in the vibration operational test program to verify the design of fluid systems.

(2) Discuss the testing procedures and analyses used in the design of Category I mechanical equipment such as fans, pumps and heat exchangers, to withstand seismic loading conditions, including the manner in which the methods and procedures to be employed will consider the frequency spectra and amplitudes calculated to exist at the equipment supports. Where tests or analyses do not include evaluation of the equipment in the operating mode, describe the bases for assuring that this equipment will function when subjected to seismic and accident loadings.

(3) The basis for the derivation of the forcing functions which will be used in the dynamic system analyses and normal reactor operation and anticipated operational transients should be specified. A brief description should be presented of the dynamic system analysis methods and procedures which will be used to determine dynamic responses of reactor internals and other Group A structures, systems, components, and equipment (e.g., analyses and tests). Discuss the verification of the dynamic system analysis by the preoperational test program, if applicable. The discussion should include the preoperational test program elements described in Safety Guide 20, Vibration Measurements on Reactor Internals. In the event elements of the program differ substantially from Safety Guide 20, the basis and justification for these differences should be presented.

(4) The preoperational testing program offers the only means to verify the reactor internals mathematical models, methods of analysis, and analytical results (e.g., ring and beam response, modes shapes, damping factors, predominate frequencies and response amplitudes). Many of the reactor internals response characteristics verified by the preoperational test are the same response characteristics that occur during the LOCA condition (e.g., mode shapes, beam and ring responses) with different response amplitudes. Provide a discussion of the preoperational analysis and testing results that will be used to augment the LOCA dynamic

analysis methods and procedures, i.e., barrel ring and beam modes, guide tube responses, water mass and compliance effects, damping factor selection, etc.

(5) The dynamic system analysis methods and procedures that were used to confirm the structural integrity of the reactor coolant system and the reactor internals under the LOCA loadings should be provided. Include a brief description of the methods, procedures, analytical and test results and sketches of the mathematical models that were used.

(6) Describe the analytical methods used to evaluate stresses (e.g., elastic or inelastic) and provide a discussion of their compatibility with the type of dynamic system analysis. Justification should be provided for the proposed use of inelastic stress analyses or application of inelastic stress limits with an elastic dynamic system analysis.

3.9.2 ASME Code Class 2 and 3 Components

The following information should be provided for all Code Class 2 and 3 components of fluid systems that are to be constructed in accordance with the ASME Boiler and Pressure Vessel Code, subsection NC and ND (or other equivalent requirements):

(1) The design pressure, temperature, and other loading conditions that provide the bases for design of systems or components should be specified.

(2) The design loading combinations (e.g., normal service or functional operating loads, seismic loads, etc.) that are considered in the component or system design should be listed.

(3) The combination of design loadings should be categorized (if applicable) with respect to either Normal, Upset, or Emergency Condition (defined in the ASME Section III Code). The stress limits associated with each of the design loading combinations should also be specified.

(4) In the event that the proposed stress limits result in inelastic deformation (or are comparable to the faulted condition limits defined in ASME Section III) provide the detailed bases for such application including a description of the methods by which it will be demonstrated that the component will maintain its functional or structural integrity under the design loading combination.

(5) Provide a list of the ASME and ANSI code case interpretations applied to all components not within the reactor coolant pressure boundary.

(6) Identify all active* pumps and valves which are not a part of the reactor coolant pressure boundary. Describe the criteria employed to assure that active components will function as designed, e.g., stress limits below yield calculated on an elastic basis (comparable to the Normal and Upset Condition limits specified in ASME Section III). Where empirical methods are employed, provide a summary description of test procedures, loading techniques and results, including the basis for extrapolations to components larger or smaller than those tested.

(7) Present the bases for the proposed design approach and the criteria used to assure the protection of all critical systems and the containment from the effects of pipe whip. For pipe breaks postulated in systems other than the reactor coolant pressure boundary provide the following information:

(a) The systems postulated to rupture

(b) Any limitations on break locations

(c) Whether both longitudinal and circumferential breaks are considered.

(8) Describe the design and installation criteria for the mounting of the pressure-relieving devices (safety valves and relief valves) on the main steam lines outside of containment for pressurized water reactors. In particular, specify the design criteria used to take into account full discharge loads (i.e., thrust, bending, torsion) imposed on valves and on connected piping in the event all the valves are required to discharge. Indicate the provisions made to accommodate these loads.

(9) List the analytical methods and criteria used to evaluate stresses and deformations in all safety related pumps and valves including safety and relief valves. For design conditions other than those explicitly addressed by the ASME Section III Code (e.g., design condition categories for which code limits have not been developed, geometries not included, etc.) provide a summary of each analytical method and the associated acceptance limits. Where empirical relationships and methods determine design, provide the bases for extrapolating these methods or experience to all loading conditions specified for each component.

---

* Active components are those whose operability is relied upon to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a postulated pipe break in the reactor coolant pressure boundary.

(10) The design conditions should be specified for all components that are to be constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, Subsections NC, ND, and NE. The design conditions provided for each component should contain the design loadings (including the design pressure, temperature, and mechanical loads), the operational cycles and the number of occurrences of each, and the design loading combinations categorized (as appropriate) with respect to the conditions identified in NA-2110 of Section III. The information submitted should include sufficient detail to provide the complete basis for the design of all classes of components intended to conform to the rules of Section III of the Code.

(11) For components that are to be constructed in accordance with Section III of the ASME Boiler and Pressure Vessel Code, Subsections NC, ND, and NE, the analytical calculations or experimental testing performed to demonstrate compliance with Section III of the Code should be provided. A complete description should be submitted of the mathematical or test models, the methods of calculation or test including any simplifying assumptions, and a summary of results which include the stresses obtained by calculation or test, cumulative damage usage factors and design margins. The information provided should be sufficiently detailed to show the validity of the structural design to sustain and meet in every respect the provisions of the Certified Design Sepcifications and the requirements of Section III of the Code.

(12) Specify the code, load combinations and stress limits for those storage tanks that are relied upon to (1) prevent or mitigate the consequences of accidents, (2) permit safe shutdown of the reactor and its maintenance in a safe shutdown condition, and (3) retain radioactive material.

3.9.3 Components Not Covered by ASME Code

For safety related mechanical components not covered by the ASME Boiler and Pressure Vessel Code, provide a summary of the stress and dynamic calculations or experimental testing performed to demonstrate that all design loading combinations will be sustained without impairment of structural integrity or functional capability. Details of the mechanical design and analytical procedures for the design of the fuel should be included (see Chapter 4.0 of the SAR).

Provide the stress and dynamic criteria, methods, and procedures which have been used to determine the operability of the control rod drives and control rod insertability under LOCA and seismic loadings. (See also Chapter 4.0 of the SAR.)

3.10 Seismic Design of Category I Instrumentation and Electrical Equipment

The seismic design criteria for the reactor protection system, engineered safety feature circuits, and the emergency power system should be provided. The criteria should address: (1) the capability to initiate a protective action during the safe shutdown earthquake, and (2) the capability of the engineered safety feature circuits and the standby power system to withstand seismic disturbances during post-accident operation. Indicate the extent of compliance with the seismic qualification procedures and documentation requirements of IEEE Std. 344-1971, "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations."

Describe the analyses, testing procedures, and seismic restraint measures employed to establish the seismic design adequacy of Category I electrical equipment supports such as cable trays, battery racks, instrument racks, and control consoles. Provide the criteria used to account for the possible amplification of the seismic floor input by the frames and racks that support electrical equipment. Include the criteria and verification procedure employed to account for the possible amplified design loads (frequency and amplitude) for vendor-supplied components.

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 assure acceptable performance in all environments (both normal and accident).

Information on 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 post-accident environment of temperature, pressure, humidity and radiation should include the following specific information:

(1) Identify 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.

(2) Describe the qualification tests and analyses that have been or will be performed on each of these items to assure that it will perform in the combined high temperature, pressure, humidity and radiation environment. Include the specific values of temperature, pressure, humidity, and radiation, noting that the accident conditions should be superimposed on the long term environment to which the equipment in question is normally

exposed. Describe and justify any exceptions to IEEE Std. 334-1971, "IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations."

(3) Provide the results of the successful completion of qualification tests for each type of equipment. In assessing the potential effects of radiation on all safety related equipment and components, use should be made of the following assumptions with respect to the fission product source term:

(a) For the purpose of calculating doses on equipment and materials, fission products assumed to be in the recirculated water should be 50% of the core halogen inventory and 1% of the core solid fission product inventory.

(b) For purposes of calculating heat loads on filters, range of radiation monitors, and radiation dose to equipment in the containment atmosphere, fission products assumed to be in the primary containment atmosphere should be 25% of the core halogen inventory, and 1% of the core solid fission product inventory.

(4) The criteria should be provided that have been established to assure that loss of the air conditioning and/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 of the limiting condition with regard to temperature that would require reactor shutdown, and how this was determined. Any testing (factory and/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.

4.0 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 throughout its design lifetime under all normal operational modes, including both transient and steady state, without releasing other than acceptably small amounts of fission products to the coolant. This chapter should also include information to support the analyses presented in Chapter 15.0, 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. A summary table of the important design and performance characteristics should be included.

4.2 Mechanical Design

4.2.1 Fuel

The design bases for the mechanical design of the fuel components should be presented including mechanical limits such as maximum allowable stresses, deflection, cycling and fatigue limits, capacity for fuel fission gas inventory, maximum internal gas pressure, material selection, radiation damage, and shock and seismic loadings. Details of the dynamic analysis, input forcing functions, vibration, and seismic response loadings should be presented in Sections 3.7 and 3.9 of the SAR.

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 upon developmental work and/or analysis should be provided. These bases may be expressed as explicit numbers or as general conditions. The discussion of design bases should include consideration of: (a) the physical properties of the cladding

and the effects of design temperature and irradiation on the properties; (b) stress-strain limits; (c) the effects of fuel swelling; (d) variations of melting point and fuel conductivity with burnup; and (e) the requirements for surveillance and testing of irradiated fuel rods.

A description and design drawings of the fuel assemblies and fuel elements showing arrangement, dimensions, critical tolerances, sealing and handling features, methods of support, fission gas spaces, burnable poison content, and internal components should be provided.

An evaluation of the fuel design should be provided including considerations such as materials adequacy throughout lifetime, a summary of results of a vibration analysis, fuel element internal pressure and cladding stresses during normal and accident conditions with particular emphasis upon temperature transients or depressurization accidents; potential for a waterlogging rupture; potential for a chemical reaction, including hydriding effects; fretting corrosion; cycling and fatigue; and dimensional stability of the fuel and critical components during design lifetime. The evaluation should include discussions of failure and burnup experience, and the thermal conditions for which the experience was obtained for the type of fuel to be used, and the results of long term irradiation testing of production fuel and test specimens.

The testing and inspections to be performed to verify the mechanical characteristics of the fuel components should be described including clad integrity, fuel pellet characteristics, radiographic inspections, destructive tests, fuel assembly dimensional checks, and the program for inspection of new fuel assemblies, new control rods, and new reactor internals to assure mechanical integrity after shipment. Where testing and inspection programs are essentially the same as for previously accepted facilities, 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.

4.2.2 Reactor Vessel Internals

The design bases for the mechanical design of the reactor vessel internal components should be presented including mechanical limits such as maximum allowable stresses, deflection, cycling and fatigue limits, fuel assembly restraints (positioning and holddown), material selection, radiation damage, and shock loadings. Details of the dynamic analyses, input forcing functions, and response loadings should be presented in Section 3.9 of the SAR.

The reactor vessel internals should be described and general assembly drawings provided showing the arrangement of the important components, positioning and support of the fuel assemblies, control rod and shim arrangement and support, and location of in-core instrumentation and reactor vessel surveillance specimen capsules.

The design loading conditions 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. All combinations of design loadings should be listed (e.g., operating pressure differences and thermal effects, seismic and transient pressure loads associated with postulated loss-of-coolant accidents) that are accounted for in design of the core support structure.

In addition, each combination of design loadings should be categorized with respect to either the Normal, Upset, Emergency or Faulted Condition (defined in the ASME Section III Code) and the associated design stress intensity or deformation limits should be stipulated.

The bases for the proposed design stress and deformation criteria should be identified (e.g., the January 1971 draft of the ASME Code for Core Support Structures - Subsection NG).

4.2.3 Reactivity Control Systems

The design bases for the mechanical design of each of the reactivity control systems should be presented including control rod clearances, mechanical insertion requirements, material selection, radiation damage, and positioning requirements. Details of the dynamic analysis and testing, stress and deformation, and fatigue limits should be discussed in Section 3.9 of the SAR.

A description of each of the reactivity control systems should be provided including design drawings of the control rods and followers, rod drives, latching mechanisms, and assembly within the reactor; design drawings and flow diagrams for chemical injection systems; and design drawings for temporary reactivity control devices for the initial core.

An evaluation of the reactivity control systems should be provided which includes considerations such as materials adequacy throughout design lifetime; results of a dimensional and tolerance analysis of the systems as a

whole, including points of support in the vessel, core structure and channels, control rods and followers, extension shafts and drive shafts; thermal analysis to determine tendencies to warp; analysis of pressure forces which could eject rods or temporary devices from the core; potential for and consequences of a functional failure of critical components; analysis of the ability to preclude excessive rates of reactivity addition; possible effect of violent fuel rod failures on control rod channel clearances; assessment of the sensitivity of the systems to mechanical damage as regards the capability to continuously provide reactivity control; and previous experience and/or developmental work with similar systems and materials.

The testing and inspections to be performed to verify the mechanical characteristics of the reactivity control systems should be described including test and surveillance programs to demonstrate proper functioning during initial start-up and throughout design lifetime.

The instrumentation to be employed in connection with mechanical and chemical reactivity control systems and reactivity monitoring should be discussed in terms of functional requirements. Details of the design and logic of the instrumentation should be discussed in Chapter 7.0 of the SAR.

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 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, maximum 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 including the following information:

(1) State the cold and hot excess reactivity and shutdown reactivity margins with and without mechanical and chemical shims and with and without equilibrium xenon and samarium poisoning, including the effects of burnable poisons, for the clean condition and the maximum reactivity condition. If different, excess reactivity associated with temperature, moderator voids, and burnup should be indicated.

(2) For hot, cold, and intermediate temperature conditions, provide the coefficients of reactivity associated with (a) moderator temperature and voids (overall and regional), (b) fuel Doppler effect, (c) fuel geometry and composition changes, and (d) fuel thermal expansion.

(3) State the hot and cold reactivity worth of individual control rods and groups of rods for planned loading patterns and core operating modes with estimates of reductions in effectiveness during core lifetime.

(4) Provide the hot and cold reactivity worth of fuel assemblies and mechanical or chemical shims.

(5) Provide the hot and cold reactivity worth of any materials within the core or adjacent to it that could have a significant reactivity effect by a change in position, as for example, flooding of superheat reactors or movement of reflecting elements, movement of temporary control devices, or flux suppression materials.

(6) Specify the maximum controlled rate of reactivity addition at startup and at operating conditions.

(7) Describe the gross and local radial and axial power distribution for different planned rod patterns with and without equilibrium xenon and samarium.

(8) Give the power decay curve for full and partial scram or power cutback, if applicable, from the least effective planned rod arrangement.

(9) State the minimum critical mass with and without xenon and samarium poisoning.

(10) Provide the neutron flux distribution and spectrum at core boundaries and at the pressure vessel wall.

(11) Indicate the expected core lifetime, and fuel burnup, and describe the fuel replacement program.

(12) Discuss the stability of the core against xenon-induced power oscillations.

4.3.3 Evaluation

An evaluation of the nuclear design should be provided. The evaluation should include a description of the analytical methods employed in arriving at important nuclear parameters, with an estimate of accuracy by comparison with experiments or with the performance of other reactors. Also included should be a discussion of the potential effects for those cases in which nuclear parameters such as excess reactivity, reactivity coefficients and reactivity insertion rates exceed prior practice. An evaluation of reactor stability should be provided.

4.3.4 Tests and Inspections

The tests and inspections necessary to verify the nuclear characteristics of the fuel and reactivity control systems should be discussed. These should include the various inspections performed during fabrication, verification of fuel enrichment, and the planned nuclear experiments and tests, both in critical assemblies and zero power and approach-to-power tests at the reactor site.

4.3.5 Instrumentation Application

This section should discuss the functional requirements for the instrumentation to be employed for monitoring and measuring core power distribution and other relevant parameters. Details of the instrumentation design and logic should be discussed in Chapter 7.0 of the SAR.

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 (at rated power, design overpower and during transients), critical heat flux

ratio (at rated power, 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

A description of the thermal and hydraulic characteristics of the reactor design should be provided including the following:

(1) Provide a summary comparison of the thermal and hydraulic design parameters of the reactor with previously approved reactors of similar design. Include, for example, primary coolant temperatures, fuel temperatures, critical heat flux ratio, and critical heat flux correlations used.

(2) Discuss and provide fuel cladding temperatures, both local and distributed, with an indication of the correlation used for thermal conductivity and the method of employing peaking factors.

(3) Provide the critical heat flux ratio, both local and distributed, with an indication of the critical heat flux correlation used, analysis techniques, method of use, method of employing peaking factors, and comparison with other correlations.

(4) Discuss the margin provided in the peaking factor employed to account for flux tilts, to assure that flux limits are not exceeded during operation.

(5) Give the predicted core average and maximum void fraction and distribution.

(6) Describe and discuss core coolant flow distribution and orificing.

(7) Provide core pressure drops and hydraulic loads during normal and accident conditions.

(8) Discuss the correlations and physical data employed in determining important characteristics such as heat transfer coefficients and pressure drop.

(9) Evaluate the capability of the core to withstand the thermal effects resulting from anticipated operational transients.

(10) Discuss 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).

(11) Provide a summary table of characteristics including important thermal and hydraulic parameters such as coolant velocities, surface heat fluxes, power density, specific power, surface areas, and flow areas.

4.4.3 Evaluation

An evaluation of the thermal and hydraulic design of the reactor should be provided including the following specific information:

(1) With respect to core hydraulics the evaluation should include:

(a) a discussion of the results of flow model tests (with respect to pressure drop for the various flow paths through the reactor and flow distributions at the core inlet); (b) the empirical correlations 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 (c) pump characteristics including consideration of NPSH requirements and conditions where all pumps are not operating.

(2) The influence of axial and radial power distributions on the thermal and hydraulic design should be discussed.

(3) The thermal response of the core should be evaluated at rated power, design overpower, and for expected transient conditions.

(4) A comprehensive discussion of the analytical techniques used in evaluating the core thermal-hydraulics should be provided, including estimates of uncertainties.

(5) Provide the results of an analysis of hydraulic instability.

(6) Provide an analysis of the potential for and effect of sudden temperature transients on waterlogged elements or elements with high internal gas pressure.

(7) Provide an analysis of temperature effects during anticipated operational transients that may cause bowing or other damage to fuel, control rods or structure.

(8) Evaluate the energy release and potential for a chemical reaction should physical burnout of fuel elements occur.

(9) Evaluate the energy release and resulting pressure pulse should waterlogged elements rupture and spill fuel into the coolant.

(10) Discuss the behavior of fuel rods in the event of coolant flow blockage.

4.4.4 Testing and Verification

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided and will remain within required limits throughout core lifetime should be discussed.

4.4.5 Instrumentation Application

This section should discuss the functional requirements for the instrumentation to be employed in monitoring and measuring those thermal-hydraulic parameters important to safety. Include, for example, the requirements for in-core instrumentation to confirm predicted power density distribution and moderator temperature distributions. Details of the instrumentation design and logic should be discussed in Chapter 7.0 of the SAR.

5.0 REACTOR COOLANT SYSTEM

This chapter of the Safety Analysis Report should provide information regarding the reactor coolant system and pressure-containing appendages out to and including isolation valving. This grouping of components is defined as the "reactor coolant pressure boundary (RCPB)", in Section 50.2(v) of 10 CFR Part 50 as follows:

> "Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:
>
> (1) Part of the reactor coolant system, or
>
> (2) Connected to the reactor coolant system, up to and including any and all of the following:
>
> (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,
>
> (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
>
> (iii) The reactor coolant system safety and relief valves.
>
> For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valve in the main steam and feedwater piping."

The portions of the system beyond the isolation valves should be treated as part of the steam and power conversion system in Chapter 10.0.

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 abnormal. 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. Provide the following specific information:

(1) 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.

(2) A piping and instrumentation diagram of the reactor coolant system and the primary sides of the auxiliary or emergency fluid systems and engineered safety feature systems interconnected with the reactor coolant system, delineating on the diagram:

(a) The extent of the systems located within the containment,

(b) The points of separation between the reactor coolant (heat transport) system and the secondary (heat utilization) system, and

(c) 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.

(3) 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 Design Criteria, Methods, and Procedures

The design criteria to be used for the components of the RCPB should be stated. They should include the following information:

(1) State the performance objectives of the system and its components from which the design parameters are derived for both the normal and transient conditions considered.

(2) State the design pressure, temperature, seismic loads, and maximum system and component test pressures for the system and individual components.

(3) Provide a Table which shows compliance with the rules of 10 CFR Part 50, Section 50.55a, "Codes and Standards." In the event there are cases wherein conformance to the rules of Section 50.55a would result in hardships or unusual difficulties without a compensating increase in the level of safety and quality, provide a complete description of the circumstances resulting in such cases and the basis for proposed alternative requirements. Demonstrate that an acceptable level of safety and quality will be provided by the proposed alternatives.

(4) Provide a list of the ASME and ANSI code case interpretations that will be applied to components within the reactor coolant pressure boundary.

(5) Provide a complete list of transients to be used in the design and fatigue analysis of all the applicable components within the reactor coolant pressure boundary discussed in Section 5.5. Specify all design transients and their number of cycles such as 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, seismic events, etc., that are contained in the ASME Code-required "Design Specifications" for the components of the reactor coolant pressure boundary. Categorize all transients or combinations of transients with respect to the conditions identified as "normal," upset," "emergency" or "faulted" as defined in the ASME Section III Nuclear Component Code. In addition, provide the design loading combinations and the associated stress or deformation limits specified. The information should include sufficient detail to provide the bases for the design of all classes of components intended to conform to the rules of Section III of the ASME Code.

(6) Provide a list which classifies pumps and valves within the reactor coolant pressure boundary as either active[1/] or inactive[2/] components. Describe the criteria employed to assure that active components will function as designed in the event of a pipe rupture (faulted condition) in the reactor coolant pressure boundary, e.g., allowable stress limits established at or near the yield stress calculated on an elastic basis. Describe the isolation signal, the closure time, and the leak-tight integrity criteria for all active valves.

[1/] Active components are those whose operability is relied upon to perform a safety function (as well as reactor shutdown function) during the transients or events considered in the respective operating condition categories.

[2/] Inactive components are those whose operability (e.g., valve opening, or closure, pump operation or trip) are not relied upon to perform the system function during the transients or events considered in the respective operating condition categories.

Where empirical methods (tests) are employed, provide a summary description of test methods, loading techniques and results including the bases for extrapolations to components larger or smaller than those tested.

(7) Provide the stress criteria associated with the emergency and faulted operation conditon categories for pumps and valves within the RCPB. If stress and pressure limits other than those specified in Paragraphs NB-3655 and NB-3656 of Section III, of the ASME Boiler and Pressure Vessel Code (1971) or ANSI B31.7 Code Case 70 are proposed for inactive components, provide the basis for their application.

(8) Specify whether the criteria to be employed in design against the effects of pipe rupture will consider pipe breaks postulated to occur at any location within the reactor coolant pressure boundary, or at limited areas within the system. Indicate whether these criteria include consideration of both longitudinal and circumferential pipe breaks and provide the bases for the design approach.

(9) The use of the plastic instability and limit analysis methods of ASME Section III may not be necessarily conservative and compatible with the type of dynamic system analysis used. Provide justification for the use of inelastic stress analysis methods in conjunction with elastic system dynamic analysis.

(10) Protection provided for the principal components of the reactor coolant system against environmental factors (e.g., fires, flooding, missiles, seismic effects) to which the system may be subjected should be discussed.

(11) For components that are to be constructed in accordance with Section III of the ASME Code, Subsection NB, the analytical calculations or experimental testing performed to demonstrate compliance with the Code should be provided. A complete description should be submitted in the FSAR of the mathematical or test models, the methods of calculation or test including any simplifying assumptions, and a summary of results which include the stresses obtained by calculation or test, cumulative damage usage factors and design margins. The information provided should be sufficiently detailed to show the validity of the structural design to sustain and meet in every respect the provisions of the Certified Design Specifications and the requirements of Section III of the ASME Code.

(12) The design stress criteria for faulted condition loadings should be specified.

(13) In the FSAR, a list of Category I systems and the associated stress levels (i.e., seismic, dead weight plus pressure, LOCA, etc.) at

all points of high changes in flexibility under the faulted condition should be provided. Include sketches of each system configuration.

(14) List the analytical methods and criteria used to evaluate stresses and deformations in all pumps and valves including safety and relief valves. For design conditions other than those explicitly addressed by the ASME Section III Code (e.g., design condition categories for which code limits have not been developed, geometries not included, etc.), provide a summary of each analytical method and the associated acceptance limits. Where empirical relationships and methods determine the design, the bases for extrapolating these methods or experience to all loading conditions specified for each component.

(15) In the PSAR, provide the methods and criteria used to preclude critical speed problems in pumps, and to confirm the integrity of the bearings for the transient conditions encountered during service.

(16) Describe the qualification test program that will be used to verify that active valves (whose operability is relied upon to perform a safety function or shut down the reactor) will operate under the transient loadings experienced during the service life.

5.2.2 Overpressurization Protection

Provide the following information regarding the provisions taken to protect the RCPB against overpressurization:

(1) Identify and show the location on P and I diagrams of all pressure-relieving devices for (a) the reactor coolant system, (b) the primary side of the auxiliary or emergency systems interconnected with the primary system, and (c) any blowdown or heat dissipation system connected to the discharge side of the pressure-relieving devices.

(2) Describe the design and installation criteria for the mounting of the pressure-relieving devices (safety valves and relief valves) within the reactor coolant pressure boundary. In particular, specify the design criteria which will be used to take into account full discharge loads (i.e., thrust, bending, torsion) imposed on valves and on connected piping in the event all the valves are required to discharge. Indicate the provisions made to accommodate these loads.

(3) To facilitate review of the bases for the pressure relieving capacity of the reactor coolant pressure boundary, submit (as an appendix to the SAR) the "Report on Overpressure Protection" that has been prepared in accordance with the requirements of the ASME Section III

Nuclear Power Plant Components Code or, if the report is not available at the time the PSAR is submitted, indicate the approximate date for submission. In the event the report is not expected to be available until either the Operating License review or late in the construction schedule for the plant, provide in the PSAR the bases and analytical approach (e.g., preliminary analyses) being utilized to establish the overpressure relieving capacity required for the reactor coolant pressure boundary.

(4) In the PSAR, describe the analytical methods used to demonstrate that the postulated occurrence of failure to scram on anticipated transients will not result in exceeding the stress limits for the Upset Condition for components of the reactor coolant pressure boundary.

5.2.3 Material Considerations

For the materials to be used in the reactor coolant pressure boundary, provide information regarding material specifications, fracture toughness requirements for ferritic steels, stress corrosion susceptibility of austenitic stainless steels, delta ferrite control in austenitic stainless steel welds, and requirements for pump flywheels. Specifically, the following information should be provided:

(1) Provide a list of specifications for the principal pressure retaining ferritic materials and austenitic stainless steels, including weld materials, intended to be used for each component (e.g., vessels, piping, pumps and valves) that is part of the reactor coolant pressure boundary. With respect to ferritic materials (including welds) of the reactor pressure vessel beltline, the information regarding these specifications should include any additionally imposed limits on residual elements (reportable and nonreportable) by specification requirements which are intended to reduce sensitivity to irradiation embrittlement in service. Any additional or special requirements by the purchaser should be indicated.

(2) Discuss the materials of construction exposed to the reactor coolant and their compatibility with the coolant and contaminants or radiolytic products to which the system may be exposed.

(3) Discuss the materials of construction of reactor coolant systems and their compatibility with external insulation or the environmental atmosphere in the event of coolant leakage.

(4) Describe the additives to be used in the reactor coolant system (such as inhibitors) whose principal function is directed toward corrosion control within the system.

(5) Describe the fracture toughness criteria specified for ferritic materials of the reactor coolant pressure boundary, and indicate the degree of compliance with the AEC proposed "Fracture Toughness Requirements," 10 CFR Part 50 Appendix G, published in the Federal Register on July 3, 1971.

(6) For all pressure-retaining ferritic components of the reactor coolant pressure boundary whose lowest pressurization temperature* will be below 250°F, provide the material toughness properties (Charpy V-notch impact test curves and dropweight test NDT temperature, or others) that have been reported or specified for plates, forgings, piping, and weld material. Specifically, for each component provide the following data for materials (plates, pipes, forgings, castings, welds) used in the construction of the component, or your estimates based on the available data:

(a) The highest of the NDT temperatures obtained from DWT tests,

(b) The highest of the temperatures corresponding to the 50 ft-lb value of the C_v fracture energy, and

(c) The lowest of the upper shelf C_v energy values for the "weak" direction (WR direction in plates) of the material.

(7) Identify the location and the type of the material (plate, forging, weld, etc.) in each component for which the data listed above were obtained. Where these fracture toughness parameters occur in more than one plate, forging or weld, provide the information requested in (3) above for each of them.

(8) For reactor vessel beltline materials, including welds, specify the highest predicted end-of-life transition temperature corresponding to the 50 ft-lb value of the Charpy V-notch fracture energy for the "weak direction" of the material (WR direction in plates), and the minimum upper shelf energy value which will be acceptable for continued reactor operation toward the end-of-service life of the vessel.

(9) List all non-stabilized grades of austenitic stainless steels (AISI Type 3XX series) with a carbon content greater than 0.03%, that will be used for components of the reactor coolant pressure boundary. In light of their susceptibility to preservice and inservice intergranular stress

*Lowest pressurization temperature of a component is the lowest temperature at which the pressure within the component exceeds 25 percent of the system normal operating pressure, or at which the rate of temperature change in the component material exceeds 50°F/hr., under normal operation, system hydrostatic tests, or transient conditions.

corrosion attack, describe the plans which will be followed to avoid partial or local severe sensitization of austenitic stainless steel during heat treatments and welding operations for core structural load bearing members and component parts of the reactor coolant pressure boundary. Describe welding methods, heat input, and the quality controls that will be employed in welding austenitic stainless steel components.

(10) To avoid microfissuring in welds, describe the requirements for control of delta ferrite in austenitic stainless steel welds, especially as regards filler materials, welding procedure qualification, and the methods for determining delta ferrite content of the completed welds.

(11) AEC General Design Criterion 4 requires that structures, systems, and components of nuclear power plants important to safety be protected against the effects of missiles that might result from equipment failures. Provide the information which demonstrates compliance with GDC-4 and AEC Safety Guide 14, relating to material properties, design, inservice inspection and testing of the reactor coolant pump flywheels.

5.2.4 RCPB Leakage Detection Systems

To demonstrate compliance with AEC General Design Criterion 30, which requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage, provide the following information:

(1) Describe the methods that will be used to determine coolant leakage from the reactor coolant pressure boundary. Provide sufficient detail to indicate that redundant systems of diverse modes of operation will be installed in the plant.

(2) Describe the methods used to provide positive indications in the control room of leakage of coolant from the reactor coolant system to the containment.

(3) Discuss the adequacy of the leakage detection system which depends on reactor coolant activity for detection of changes in leakage during the initial period of plant operation when the coolant activity may be low.

(4) With reference to the proposed maximum allowable leakage rate from unidentified sources in the reactor coolant pressure boundary, furnish the following information:

(a) The length of a through-wall crack that would leak at the rate of the proposed limit as a function of wall thickness.

(b) The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.

(c) The mathematical model and data used in such analyses.

(5) Specify the proposed maximum allowable total leakage rate for the reactor coolant pressure boundary, and the basis for the proposed limit. Furnish the ratio of the proposed limit to the normal capacity of the reactor coolant makeup system, and to the normal capacity of the containment water removal system.

(6) Provide the sensitivity (in gpm) and the response time of each leak detection system. For the containment air activity monitors, provide the sensitivity and the response time as a function of the percentage of failed fuel rods or of the corrosion product activity in the reactor coolant, as applicable.

(7) Estimate the anticipated normal total leakage rates and major leakage sources on the basis of operational experience from other plants of similar design.

(8) Describe the adequacy of the proposed leakage detection systems to differentiate between identified and unidentified leaks from components within the primary reactor containment and indicate which of these systems provide a means for locating the general area of a leak.

(9) Discuss the criteria for shutdown of the reactor in the event that either the total or unidentified leakage rate limit was exceeded.

(10) Describe the tests proposed to demonstrate sensitivities and operability of the leakage detection systems.

5.2.5 Inservice Inspection Program

To demonstrate compliance with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems", provide the following information:

(1) Describe the design and arrangement provisions for access to the reactor coolant pressure boundary as required by Section IS-141 and IS-142 of Section XI of the ASME Boiler and Pressure Vessel Code - Inservice Inspection of Nuclear Reactor Coolant Systems. Indicate the specific provisions made for access to the reactor vessel for examination of the welds and other components.

(2) Section XI of the ASME Boiler and Pressure Vessel Code recognizes the problems of examining radioactive areas where access by personnel will be impractical, and provisions are incorporated in the rules for the examination of such areas by remote means. In some cases the equipment to be used to perform such examination is under development. Provide the following information with respect to your inspection program:

(a) Describe the equipment that will be used, or is under development for use, in performing the reactor vessel and nozzle inservice inspections.

(b) Describe the system to be used to record and compare the data from the baseline inspection with the data that will be obtained from subsequent inservice inspections.

(c) Describe the procedures to be followed to coordinate the development of the remote inservice inspection equipment with the access provisions for inservice inspection afforded by the plant design.

(3) Describe plans for inservice monitoring of the reactor coolant system for the presence of loose parts and excessive vibration.

5.3 Thermal Hydraulic System Design

The thermal hydraulic design of the reactor coolant system should be described in this section. The following specific information should be included:

(1) State the bases for design of the system (e.g., the linear heat generation rates and the critical heat flux ratio for both transient and steady-state conditions).

(2) State the core peaking factors and explain the basis for their selection as a function of fuel exposure.

(3) Describe the analytical methods, thermodynamic data, and hydrodynamics data used to determine the thermal and hydraulic characteristics of the reactor coolant system.

(4) State the operating restrictions that will be imposed on the coolant pumps to meet net positive suction head requirements.

(5) For boiling water reactors, provide a power-flow operating map indicating the limits of reactor coolant systems operation. This map should indicate the permissible operating range as bounded by minimum flow, design flow, maximum pump speed, and natural circulation.

(6) For pressurized water reactors, provide a temperature-power operating map indicating the effects of reduced core flow due to inoperative pumps including system capability during natural circulation conditions.

(7) Describe the load following characteristics of the reactor coolant system and the techniques employed to provide this capability.

(8) Discuss the transient effects of such events as loss of full or partial coolant flow, coolant pump speed changes, load changes, and start-up of an inactive loop.

(9) Provide a table summarizing the thermal and hydraulic characteristics of the reactor coolant system.

5.4 Reactor Vessel and Appurtenances

The discussion in this section should present the design bases, description, evaluation, and necessary tests and inspections for the reactor vessel and its appurtenances.

The following specific information should be provided as a minimum:

(1) Specify the maximum normal and emergency heating and cooling rates that will be imposed on the reactor vessel to limit thermal loadings to within design specifications.

(2) Describe the extent to which the design of affected systems and components has been reviewed to determine that annealing of the reactor pressure vessel will be feasible, should it be necessary because of radiation embrittlement after several years of operation. State the maximum reactor vessel temperature that can be obtained using an in-place annealing procedure.

(3) Describe the reactor vessel material surveillance program to indicate the degree of compliance with the AEC proposed "Reactor Vessel Material Surveillance Program Requirements," 10 CFR Part 50, Appendix H, published in the Federal Register on July 3, 1971. State also the degree of conformance with ASTM E-185-70, especially with regard to the requirements on retention of representative test stock (archive material) and documentation of chemical composition.

(4) Identify and discuss any special processes to be used for the fabrication and inspection of the vessel.

(5) Describe any special design and fabrication features incorporated in the vessel to further improve its reliability and reduce its potential for failure.

(6) Identify the reactor vessel fabricator and the extent of quality assurance surveillance to be provided by the applicant or his representative (particularly if the vessel is to be fabricated outside the U.S.).

(7) Discuss reactor vessel lifetime design transients in terms of number of cycles anticipated for each type of transient.

(8) State the vessel materials and inspections to be carried out during fabrication.

(9) Provide reactor vessel design data in tabular form.

5.5 Component and Subsystem Design

This section should present discussions of the performance requirements and design features to assure 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 5.5.1 through 5.5.x) for each principal component or subsystem. The discussion in each subsystem should present the design bases, description, evaluation, and necessary tests and inspections for the component or subsystem. 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.

(1) Reactor Coolant Pumps - In addition to the discussions of design bases, description, evaluations, and tests and inspections, discuss the provisions taken to preclude turbining of the reactor coolant pumps in the event of a design basis LOCA.

(2) Steam Generators - The information provided should include estimates of the radioactivity levels anticipated in the secondary side of the steam generators during normal operation, and the bases for the estimate. The potential effects of tube ruptures should be discussed.

Provide the steam generator design criteria employed to assure that flow induced vibration and cavitation effects will not result in degradation of the primary or secondary side, due to tube thinning and corrosion and erosion mechanisms, during the service lifetime of the equipment. Include the following specific information:

(a) Identify the design conditions and transients which will be specified in the design of the steam generator tubes, and the operating condition category selected (e.g., upset, emergency, or faulted) which defines the allowable stress intensity limits to be used. Justify the basis for the selected operating condition category.

(b) Specify the margin of tube-wall thinning which could be tolerated without exceeding the allowable stress limits identified in (a) above, under the postulated condition of a design basis pipe break in the reactor coolant pressure boundary during reactor operation.

(c) Describe the inservice inspection which will be employed to examine the integrity of steam generator tubes as a means to detect tube-wall thinning beyond acceptable limits and whether excess material will intentionally be provided in the tube wall thickness to accommodate the estimated degradation of tubes during the service lifetime.

(3) Reactor Coolant Piping - The subsection 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. The discussion should include the provisions taken during design, fabrication and operation to control those factors that contribute to stress corrosion cracking. Describe the provisions made for inservice inspection of the reactor coolant piping and associated components.

(4) Main Steam Line Flow Restrictors

(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.

(6) Reactor Core Isolation Cooling System

(7) Residual Heat Removal System - The radiological considerations of the residual heat removal system from a viewpoint of how radiation affects the operation of the components and from a viewpoint of how radiation levels affect the operators and capabilities of operation

and maintenance should be summarized here and derived and justified in Chapter 12.

(8) Reactor Coolant Cleanup System - The radiological considerations of the reactor coolant cleanup system should be summarized here and derived and justified in Chapters 11 and 12.

(9) Main Steam Line and Feed Water Piping

(10) Pressurizer

(11) Pressurizer Relief Tank

(12) Valves

(13) Safety and Relief Valves

(14) Component Supports

5.6 Instrumentation Application

The instrumentation to be provided in connection with the reactor coolant system and its appendages should be discussed with respect to functional requirements. Details of the design and logic of the instrumentation should be discussed in Chapter 7.0.

6.0 ENGINEERED SAFETY FEATURES

Engineered safety features are provided to mitigate the consequences of postulated serious accidents, in spite of the fact that these accidents are very unlikely. This chapter of the SAR should present information on the engineered safety features provided in the proposed plant. The information provided should be directed primarily toward showing that:

(1) the concept upon which the operation of the system is predicated has been, or will be, proven sufficiently by experience, tests under simulated accident conditions, or conservative extrapolations from present knowledge;

(2) the system will function during the period required and will actually accomplish its intended purpose;

(3) the system will function when required and will continue to function for the period required (e.g., include consideration of component reliability, system interdependency, redundancy and separation of components or portions of system); and

(4) provisions have been made for test, inspection, and surveillance and suitable testing and inspection will be performed periodically to assure that the system will be dependable and effective upon demand.

The engineered safety features included in reactor plant designs vary from facility to facility. The engineered safety features explicitly discussed in the sections of this chapter are those that are commonly used to limit the consequences of postulated accidents in light water-cooled power reactors. They should be treated as illustrative of the engineered safety features that should be treated in this chapter of the SAR, and of the kind of informative material that is needed. Where additional or different types of engineered safety features are used, they should be covered in a similar manner in separate added sections (see Section 6.X).

6.1 General

This section should identify and provide a brief summary of the types of engineered safety features provided in the plant. List each system of the plant that is considered to be an engineered safety feature.

6.2 Containment Systems

This section of the Safety Analysis Report should provide information in sufficient detail to permit the regulatory staff to evaluate the performance capability of the facility containment system. Structural design criteria

for the containment system should be provided in Chapter 3. The containment system is considered as composed of the containment structure or structures (e.g., secondary containment or confinement building) and the directly associated systems upon which the containment function depends (e.g., the system of isolation valves installed to maintain or re-establish containment system integrity when required, and the filtered ventilation system of a double or secondary containment).

In the design of nuclear power plants, the containment system which encloses the reactor and other portions of the plant constitutes a design feature provided primarily for the protection of public health and safety. Being a standby safety system, it may never be called upon to function, but it must be maintained in a state of readiness. The ability to perform its intended role, if called upon, of acting as a barrier to confine potential releases of radioactivity from severe accidents, depends upon maintaining tightness within specified bounds throughout its operating lifetime.

The SAR should include information to show that the containment system has been evaluated to provide assurance that the containment will fulfill its intended objectives, and that such objectives are consistent with protection of the public safety.

Information provided should permit a determination of the adequacy of the evaluations; that is, assurance that the evaluations included are correct and complete. Evaluations in other sections having a bearing on the adequacy of the containment system should be referenced.

6.2.1 Containment Functional Design

6.2.1.1 Design Bases - This section should provide the bases upon which the functional design of the containment system (or systems) was established, including, for example, the following information:

(1) The postulated accident conditions and the extent of simultaneous occurrences that determine the containment design requirements should be discussed.

(2) The assumptions regarding the sources and amounts of energy and material that might be released into the containment structure, and the post-accident time-dependency associated with these releases should be presented and discussed.

(3) The assumed contribution of other engineered safety features in limiting the maximum value of the energy released in the containment structure in the event of an accident should be specified.

(4) Discuss subcompartment differential pressure considerations and capability including the theoretical mass and energy input that might result from design basis accidents, particularly for those vital subcompartments that can not be pressure tested. (The structural design of the vital subcompartments with respect to accommodating this mass and energy input should be discussed in Section 3.8.2.)

(5) Discuss parameters affecting the assumed capability for post-accident pressure reduction.

6.2.1.2 System Design - This section should provide a discussion of the design features of the containment system (or systems) and the explanation* for their selections, including, for example: (a) the design internal pressure, temperature, and volume; (b) the design basis accident leakage rate, and other leakage rates as defined in the proposed Appendix J to 10 CFR Part 50; and (c) the design methods that will be used to assure integrity of the containment internal structures and subcompartments from pressure pulses that could occur following a loss-of-coolant accident.

6.2.1.3 Design Evaluation - Provide a comprehensive discussion of the evaluations** of operational systems associated with the containment which serve to indicate or maintain the state of readiness of the containment within a specified leakage rate limit during operating periods when containment integrity is required, including, for example the following information:

(1) Discuss the extent to which assurance of containment leak-tightness at any time depends upon the operation of a system, such as a continuous leakage monitoring system, a continuous leakage surveillance system, a continuous leakage surveillance system for containment penetrations and seals or a pumpback compressor system or ventilation system which maintains a negative pressure between dual barriers of a containment system.

(2) Provide an analysis of the capability of these operational systems to perform their functions reliably and accurately during operating periods and under conditions of operating interruptions (e.g., the performance margin, if any, in a pumpback compressor system that might allow it to sustain an operational failure and still function adequately).

---

*Where an explanation is given in other sections, only cross referencing is necessary.

**Where safety analyses and the discussion of the consequences of accidents under which the containment function becomes essential are included in chapter 15.0, "Accident Analysis," only cross referencing is necessary.

(3) Provide containment pressure transient analysis to establish the performance capability for a spectrum of reactor coolant break sizes up to and including rupture of the largest pipe in the primary coolant pressure boundary. Where confirmatory tests have been performed to demonstrate the applicability of the analysis, the types of tests and the results should be discussed.

(4) Describe the analytical mode, including assumptions and the methods used to verify the correctness of the mathematical formulation, and the applicability of the model to the plant design.

(5) For pressure reduction containment concepts, the effects of steam bypass on the capability of the containment to perform its design function for a complete spectrum of primary coolant break sizes should be discussed and substantiated through analyses or tests.

(6) Evaluate the long-term performance of the containment upon completion of blowdown and initial depressurization of the containment. Describe the capability of the containment systems to maintain low long-term pressure levels. Describe the analytical model, the assumptions used, the validity of the model and the results.

(7) For the design basis loss-of-coolant accident, provide an accident chronology to indicate the time of occurrence in seconds (assuming time equals zero is when the design break occurs) of events such as: initiation of the ECCS injection phase, the time containment reaches peak pressure, the end of blowdown, the end of the injection phase, initiation of the ECCS reflooding phase (assuming no offsite power), initiation of the quench containment spray, the time at which the refueling water storage tank (or condensate storage tank) empties, and where applicable, when the containment pressure becomes subatmospheric.

(8) Provide an energy balance table that lists how the energy is stored prior to the design basis loss-of-coolant accident, how much energy is generated and absorbed from time equals zero to the time of the peak pressure, and how the energy is distributed at the time of the peak pressure.

(9) Assuming a design basis loss-of-coolant accident and minimum engineered safety feature performance, and considering a time scale commencing just prior to and continuing for at least one day into the recirculation phase, provide curves showing the behavior as a function of time of: the sump temperature, the heat generation rate from core decay heat and other sources (e.g., hot metal and structures), the heat removal rate from the containment spray system heat exchanger, from the fan recirculation heat exchanger, and from the residual heat removal heat exchanger, and the containment total pressure, vapor pressure and temperature.

(10) Where applicable, with respect to the containment subcompartments enclosing such components as the reactor vessel (reactor cavity), the pressurizer, and steam generators, provide the assumptions and results of analyses to show the theoretical capability of these compartments to withstand energy releases (expressed in terms of equivalent pipe rupture area - or other applicable unit) that might result from design basis accidents. The structural design aspects of the subcompartments should be discussed in Section 3.8.2.

6.2.1.4 Testing and Inspection - This section should provide information about the program of testing and inspection applicable to: (1) preoperational testing of the containment system, and (2) in-service surveillance to assure continued integrity.

Emphasis should be given to those tests and inspections considered essential to a determination that performance objectives have been achieved and a performance capability maintained throughout the plant lifetime above some pre-established limits. Such tests could include for example: integrated leak rate tests of the containment structure, local leak detection tests of penetrations and valves and operability tests of fail-safe features of isolation valves. The information provided in this section should include, for example:

(1) the planned tests and their purpose;

(2) the considerations that led to periodic testing and the selected test frequency;

(3) the test methods to be used, including a sensitivity analysis;

(4) the requirements for acceptability of observed performance and the bases for them;

(5) the action to be taken in the event acceptability requirements are not met;

(6) information to show the extent of conformance to proposed Appendix J of 10 CFR Part 50, "Reactor Containment Leakage Testing of Water Cooled Power Reactors", published in the Federal Register on August 27, 1971; and

(7) a discussion of the design provisions to assure that the containment structure will have the capability of being pressurized to the calculated peak accident pressure at any time during plant life in order to perform integrated leakage rate tests, as may be required.

Particular emphasis should be given to those surveillance type tests that are of such importance to safety that they may become a part of the technical specifications of an operating license. The bases for such surveillance requirements should be described.

6.2.1.5 Instrumentation Application - This section should discuss the instrumentation to be employed for monitoring the containment system and actuating those components and subsystems of the containment system that initiate the safety function. Design details and logic of the instrumentation should be discussed in Chapter 7.0 of the SAR.

6.2.2 Containment Heat Removal Systems

The components and the systems for heat removal following blowdown from a loss-of-coolant accident under post-accident conditions should be considered in this section. Since the components and systems vary depending on reactor type and plant, the information to be included in this section as outlined below is only illustrative of the type of information that should be provided for each component or system.

6.2.2.1 Design Bases - Provide the bases upon which the design of the heat removal components and systems were established including, for example: (1) the sources and amounts of energy that must be considered in sizing the heat removal systems, (2) the extent to which the operation of the heat removal systems is relied upon to attenuate the post-accident conditions imposed upon the containment system, and (3) the design parameters for the portions of the heat removal systems located outside the containment.

6.2.2.2 System Design - The design features of the heat removal systems (e.g., containment spray system or fan cooler systems) should be provided in this section including, for example: (1) a description of the components and system; (2) the design specifications for the components and systems (e.g., design head of pumps, flow rate, heat removal capacity and other pertinent specifications) with adequate backup information to demonstrate that systems designed to these specifications can perform their intended function; (3) material compatibility, particularly for those systems in contact with borated water or water with other chemical additives; (4) the requirements for redundancy and independence of the components and systems; (5) the design of the recirculation piping leading from the containment sump to the recirculation pumps (e.g., the residual or decay heat removal pumps) and the means provided to detect and further reduce the potential for containment and component leakage as a possible result of component deterioration during the post-accident recirculation period (e.g., use of guard pipes surrounding the recirculation piping and the protective chambers enclosing the isolation valves); (6) the net positive suction head requirements for the recirculation pumps with supportive

information to show the margin between the required and available net positive suction head (see AEC Safety Guide No. 1); (7) consideration given to the potential for surface fouling of the containment spray system heat exchangers in the design, and the manner in which such fouling could affect the performance requirements; and (8) with respect to the containment spray and/or residual heat removal system heat exchangers, the basis for the selection of the tube side and shell side inlet temperatures and the effect on performance of the heat removal capability of the containment spray system.

6.2.2.3 Design Evaluation - This section should provide evaluations* of the heat removal systems. A description should be provided of the analytical methods and models used to assess the performance capability of the heat removal systems with sufficient information to show the validity of the models (e.g., results of tests). Summarize the results of failure analyses for all components of the heat removal systems to show that the failure of any single component will not prevent fulfilling the design function. Provide curves showing the calculated performance of the following variables as functions of time following occurrence of a design basis loss-of-coolant accident, assuming minimum engineered safety feature performance (cover a time range beginning just prior to, and continuing for at least one day into, the recirculation phase): sump temperature, heat generation rate from core decay heat and other sources (e.g., hot metal and structures), heat removal rate from the containment spray system heat exchanger, from the fan recirculation system heat exchanger, and from the residual heat removal heat exchanger, and the containment total pressure, vapor pressure and temperature.

6.2.2.4 Testing and Inspections - This section should describe the preoperational performance tests and in-place testing after installation of the heat removal systems. The description should make clear the scope and limitation of the tests. This section should also describe the inspection program for the systems, particularly for those components which will be unable to be tested after installation or periodically during operation.

6.2.2.5 Instrumentation Application - This section should describe the instrumentation to be employed for the monitoring and actuation of the containment heat removal systems. Details of the design and logic of the instrumentation should be discussed in Chapter 7.0 of the SAR.

6.2.3 Containment Air Purification and Cleanup Systems

The systems for ventilation of the containment systems (including secondary or confinement buildings) and for other air purification or cleanup systems (e.g., containment spray system and internal and external filters) servicing

* Where safety analyses and the discussion of the consequences of accidents under which the containment function becomes essential are included in chapter 15. "Accident Analyses," only cross referencing is necessary.

the containment systems should be considered as part of the containment system and discussed in this section of the SAR. (Reference should be made to Chapter 15.0, "Accident Analyses", where these containment functions become essential in describing the consequences of accidents..) The type of information outlined below should be provided for each of the cleanup systems.

6.2.3.1 Design Bases - This section should provide the design bases for the ventilation and the air purification systems, including, for example: (1) the conditions which establish the need for ventilation or purging of the containment structure, (2) the bases employed for sizing the ventilation, purging, and air cleanup systems and components, and (3) the bases for the fission product removal capability and component sizing of the containment spray system and/or filtration system (where credit is taken for limiting the radiological offsite consequences resulting from the accidents discussed in Chapter 15.0 of the SAR).

6.2.3.2 System Design - This section should discuss the design features and fission product removal capability of the systems, including, for example: (1) piping and instrumentation diagrams of the ventilation and other cleanup systems; (2) performance objectives (e.g., ventilation flow rates, temperature, humidity, the limits of radioactivity levels to be maintained within the containment structure, and at the site boundary and exclusion zone); and (3) provisions to exhaust, monitor, and filter the ventilation and purging air and the provisions for safe disposal of the effluent to the outside atmosphere (e.g., systems discharging the effluent through stacks). The following specific information should be included.

(1) The description of external charcoal filter systems should include flow parameters; charcoal type, weight, distribution, test specifications, and acceptance criteria; HEPA filter type and specifications; any additional components; humidity controls; system test and surveillance requirements; and expected efficiencies for iodine removal for each of the expected forms of iodine. Except for humidity control, the same information as above should be included in describing the internal charcoal filter systems, and in addition pressure surge data should be included.

(2) Where building recirculation systems are provided the system description should include a discussion of the mode(s) of operation and mixing behavior. Layout drawings of system equipment and air flow guidance ducts should be provided. Provide the expected initial and final exhaust flow rates and the rate of change between initial and final flow rates; the recirculation rate; and the mixing volume. If charcoal filters are included in the system, information similar to that noted in the preceding paragraph should be provided.

(3) For redundant emergency ventilation systems containing charcoal filters, describe and evaluate the design provisions for maintaining a flow of cooling air in the isolated filter train or for alternate cooling to preclude substantial fission product desorption or ignition of the charcoal (assuming a filter failure or fire occurs subsequent to a design basis accident). In the evaluation, assume the filter contains the maximum decay heat load, using as a basis the source terms indicated in Safety Guide No. 4 for Pressurized Water Reactors and Safety Guide No. 3 for Boiling Water Reactors.

(4) The important system parameters of the containment spray system that should be described and justified include flow rate through the spray nozzles, fall height (area averaged), effective containment volume and fractional volume spray coverage, the type(s) of nozzles and associated spray drop size spectrum, and also the type of spray additive along with its concentration in storage and during and following delivery.

(5) With respect to materials compatibility, an inventory should be provided of all materials which may adversely affect, or be adversely affected by, the spray solution during storage or under post-accident conditions. The system description should include a discussion of the operating modes, reliability, reproducibility, and testability of the spray system.

6.2.3.3 Design Evaluation - This section should provide evaluations of the ventilation and cleanup systems to demonstrate their capability to reduce accident doses and maintain offsite effluent concentrations during normal operation within established guidelines.

6.2.3.4 Tests and Inspections - This section should provide information concerning the program of testing and inspection applicable to preoperational testing and in-service surveillance to assure a continued state of readiness to perform for those ventilation and cleanup systems required to reduce the radiological consequences of an accident.

6.2.3.5 Instrumentation Application - This section should describe the instrumentation to be employed for the monitoring and actuation of the ventilation and cleanup systems. Design details and logic of the instrumentation should be discussed in Chapter 7.0 of the SAR.

6.2.4 Containment Isolation Systems

The system intended to monitor the development of gross leakages or measurement of leakages within allowable limits in the containment system (leakage pumpback systems which monitor containment barrier leakages may be included under this category) should be considered as part of the containment system.

The following type of information should be included:

6.2.4.1 Design Bases - Discuss the bases established for the design of the isolation valving required for fluid lines, including, for example:

(1) the governing conditions under which containment isolation becomes mandatory;

(2) the criteria applied with respect to the number and location (inside or outside of containment) of independent isolation valves provided for each fluid system penetrating the containment and the basis therof, and the degree of conformance to criteria 54, 55, 56 and 57 of the AEC General Design Criteria; and

(3) the design bases for isolation of the fluid instrument lines and the degree of conformance to AEC Safety Guide 11 or other criteria that provide an equivalent degree of protection.

6.2.4.2 System Design - Describe and evaluate the design features of the isolation valving system, including, for example:

(1) a piping and instrumentation diagram of the isolation valving system indicating the location with respect to the containment barrier of all isolation valves and fluid systems penetrating the containment wall, including instrument lines, or systems communicating directly with the outside atmosphere, (e.g., vacuum relief valves);

(2) a summary table of the types of isolation valves provided, including: (a) open or closed status under normal operating conditions, shutdown or accident situations; (b) the primary and secondary modes of actuation provided for the isolation valves, (e.g., valve operators, manual remote or automatic); (c) the number of parameters sensed and their values which are required to effect closure of isolation valves; and (d) the closure time and sequence of timing for the principal isolation valves to secure containment isolation;

(3) the protection to be provided for isolation valves, actuators, and controls against damage from missiles;

(4) the provisions to assure operability of isolation valve systems under accident environment, (e.g., imposed pressures and temperatures of the steam-laden atmosphere in the event of an accident);

(5) the provisions to assure integrity of the isolation valve system and connecting lines under the dynamic forces resulting from inadvertent closure under operating conditions (e.g., inadvertent closure of steamline isolation valves under full steaming rate); and

(6) the design of isolation valves not discussed in other sections of the SAR.

6.2.4.3 Design Evaluation - Provide an evaluation of the containment isolation system to demonstrate its capability to perform its intended function.

6.2.4.4 Tests and Inspections - Provide information concerning the program of testing and inspection that is required to assure a continued state of readiness of the system to perform its safety function.

6.2.5 Combustible Gas Control in Containment

General Design Criterion 41 requires that systems to control hydrogen, oxygen, and other substances that may be released into the reactor containment be provided as necessary to control their concentrations following postulated accidents to assure that containment integrity is maintained. This subsection of the report should provide information on the design features to be provided for controlling combustible gas concentrations in containment following an accident.

6.2.5.1 Design Bases - Discuss the bases for the design of the system and components provided to control combustible gas mixtures in the containment following a design basis loss-of-coolant accident, including, for example:

(1) the design criteria as compared to those set forth in AEC Safety Guide No. 7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident;"

(2) the design criteria applicable to the containment purge system as a backup system for the control of combustible gases in containment following an accident; and

(3) the governing conditions under which containment combustible gas control measures become necessary.

6.2.5.2 System Design - Describe the design features of the combustible gas control system, including, for example:

(1) a piping and instrumentation diagram of the system delineating the extent of the system located inside or outside containment;

(2) the concept upon which the operation of the system is predicated;

(3) the design features of the systems for mixing, sampling, and monitoring the containment atmosphere to effect control of combustible gases following a loss-of-coolant accident; and

(4) the requirements for redundancy and independence and the interdependency between the system and other engineered safety features.

6.2.5.3 Design Evaluation - Provide evaluations to demonstrate the functional requirements of the system. Provide an analysis of hydrogen generation following a loss-of-coolant accident using the assumptions set forth in AEC Safety Guide No. 7, and an analysis of the predicted thyroid and whole body doses at the site boundary and the low population zone boundary that would result from containment purging in the event of a design basis loss-of-coolant accident, using the assumptions set forth in Safety Guides No. 3 or 4, as applicable to the plant site, and Safety Guide No. 7.

6.2.5.4 Testing and Inspections - The preoperational performance tests and in place testing after installation should be described. The description should make clear the scope and limitation of the tests. Describe the inspection program for the system, particularly if the system or significant components are not testable after installation or periodically during operation.

6.2.5.5 Instrumentation Application - Discuss the instrumentation provisions for the methods of actuation (e.g., automatic, manual, different locations). The conditions requiring system actuation and the bases for the selection should be included. The design details and logic of the instrumentation should be discussed in Chapter 7.

6.3 Emergency Core Cooling System

The emergency core cooling system (ECCS) is included in a facility to furnish cooling water to the core to compensate for loss of normal cooling capability inherent in postulated loss-of-coolant accidents.

The ECCS generally consists of subsystems for storing sources of water, delivering and distributing coolant to the core, removal of heat following flow through the core, and the associated instrumentation.

The specific design requirements of an emergency core cooling system will depend upon the reactor design. Such matters as the time available following coolant loss, cooling capacity required, and the length of time during which cooling must be sustained vary. The functional requirements for the system and an explanation of why these were established should be a fundamental part of this section of the SAR.

When discussing the factors of dependability and effectiveness, specific attention should be directed to such things as system starting, adequate coolant delivery, availability of coolant, period of time the system must operate, the effect of external forces, the state of the art and proposed research and development to assure proper flow distribution to adequately cool the core, the testing program to assure dependable operation, and the reliance placed on the system for overall plant safety.

The ability of the system to start and to deliver the required cooling capacit is fundamental. Considerations should include the design, operation, and testing that are associated with system dependability from the sensing of an accident, through the availability of emergency power, to the assurance of adequate coolant flow in the core.

The potential for damage to the system from external forces, such as missiles and forces causing movement or vibration should be evaluated; e.g., since parts of the ECCS are connected to the main coolant system, assurance should be provided that an accidental rupture of the main coolant piping system will not cause movement that would negate or reduce the effectiveness of the emergency core cooling system.

Evaluations to show that there will be adequate and proper flow distribution through the core are important. Such matters as the number of channels, the effect of channel length, the phase change of the cooling water, potential metal-water reactions, and the lag time associated with system operation should be considered.

On June 19, 1971, the AEC issued an Interim Policy Statement containing interim acceptance criteria for the performance of emergency core cooling systems in light-water nuclear power plants. The Statement and an Amendment issued December 18, 1971, also included a description of acceptable assumptions and analytical procedures to be used in evaluating the performance of emergency core cooling systems for pressurized water reactors and boiling water reactors (evaluation models). The performance evaluations included in the Safety Analysis Report should be conducted in accordance with the Interim Policy Statement, and amendments thereto.

Since the system does not operate in its entirety except following an accident, a measure of its dependability must be assured through testing.

Information concerning the proposed initial tests and subsequent periodic tests and inspections should be included.

The following subsections identify information that should be included in this section.

6.3.1 Design Bases

The design of the ECCS is based upon the assumption of an accidental pipe break in the primary coolant system and the manner in which this might affect the core, and the environment in which the system will operate. The ability of a system to satisfactorily accommodate a break of a certain size does not necessarily mean it can accommodate all breaks. Therefore, the bases for setting the functional requirements of the ECCS should be identified and explained. The design bases should include, for example:

(1) the range of reactor coolant system ruptures and coolant leaks (from the smallest, up to and including the double ended rupture of the largest pipe in the reactor coolant system) that the ECCS (and subsystems) was designed to accommodate and the analyses* supporting the selection;

(2) the fission product decay heat that the ECCS was designed to remove and the analyses* supporting this selection;

(3) the reactivity required for cold shutdown for which the ECCS was designed and the analyses* supporting this selection; and

(4) the system capability to meet functional requirements over both the short and long term duration of the accident including specific features (e.g., a switch over to different coolant delivery paths) provided to meet such requirements.

6.3.2 System Design

This section should describe how the ECCS has been designed to meet the functional requirements established from the safety analyses. The information on an emergency core cooling system should include the following specific items:

(1) Provide schematic piping and instrumentation diagrams of the system showing the location of all components, piping, storage facilities, points where connecting systems and subsystems tie together and into the reactor system, and instrumentation and controls associated with subsystem and component actuation.

---

* Where these analyses have been made in other section, e.g., in Chapter 15.0, "Accident Analyses," only cross referencing is necessary.

(2) Equipment and components installed to satisfy the functional requirements should be described. Identify the significant design parameters for each component within the system. For the range of pipe-break sizes considered in the design of the ECCS, specify the components required and demonstrate that adequate coverage of the break spectrum is achieved.

(3) Identify the industry codes and classifications used in system design. Cross referencing may be used where this is discussed in other sections of the SAR.

(4) Identify the materials used in the ECCS and discuss materials compatibility.

(5) State the design pressure and temperature of components for various portions of the system and explain the bases for their selection.

(6) State the capacity of each of the coolant storage facilities.

(7) Provide pump characteristic curves and pump power requirements.

(8) Describe the heat exchanger characteristics including design flow rates, inlet and outlet temperatures for the cooling fluid and the fluid being cooled, the overall heat transfer coefficient and the heat transfer area.

(9) Provide flow diagrams for the ECCS, showing flow rates and pressure for various operating modes (i.e., emergency, test and faulted conditions).

(10) State the relief valve capacity and settings or venting provisions included in the system.

(11) Discuss the reliability considerations incorporated in the design to assure the system will start when needed and will deliver the required quantity of coolant (e.g., redundancy and separation of components, transmission lines, and power sources). A distinction should be made between true redundancy incorporated in a system and multiple components (e.g., a system that is designed to perform its function with only one of two pumps operating has increased reliability by redundancy; whereas, a system that has two pumps both of which must operate to perform its function does not have redundancy).

(12) Describe the provisions taken to protect the system (including connections to the reactor coolant system or other connecting systems) against damage that might result from movement (between components within the system and connecting systems), from missiles, or from thermal stresses.

(13) Describe the provisions taken to facilitate performance testing of components (e.g., bypasses around pumps, sampling lines, etc.).

(14) Specify the available and required net positive suction head for the ECCS pumps and justify any exceptions to the regulatory position stated in AEC Safety Guide No. 1.

(15) For PWRs, describe the provisions with respect to the control circuits for the motor-operated isolation valves in the lines connecting the ECCS accumulators (or core flooding tanks) to the reactor coolant system to preclude inadvertent closure prior to or during an accident. It should be stated whether the design of the controls for these valves will meet the intent of IEEE Std. 279-1971, "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations," and whether the following features are incorporated:

(a) automatic opening of the valves when the reactor coolant system pressure exceeds a preselected value (specified in Technical Specifications) or a safety injection signal has been initiated;

(b) valve position visual indication that is actuated by sensors on the valve ("open" and "closed");

(c) an audible alarm, independent of item (b) which is actuated by a sensor on the valve when the valve is not in the fully open position; and

(d) utilization of a safety injection signal to automatically remove (override) any bypass feature that may be provided to allow a motor-operated valve to be closed, for short periods of time, when the reactor coolant system is at pressure (in accordance with the provisions of the Technical Specifications).

(16) Describe the provisions taken in the design of the control circuits for the motor-operated isolation valves in the letdown line connecting the reactor coolant system to the relatively low pressure shutdown heat (decay or residual) removal system to preclude over-pressurization of the shutdown heat removal system as a result of common mode failures or operator errors. State whether the design of the controls for these valves will incorporate the following features:

(a) provision of at least two valves, in series, with each valve interlocked to prevent valve opening unless the reactor coolant system pressure is less than the design pressure of the shutdown heat removal system;

(b) interlocks of diverse principles, and designed to meet the intent of IEEE-279; and

(c) provision for automatic closure of the two series valves whenever the pressure in the reactor coolant system exceeds a selected fraction of the design pressure of the shutdown heat removal system. Indicate whether these closure devices will be designed to meet the intent of IEEE-279.

6.3.3. Performance Evaluation

The functional requirements established for the emergency core cooling system generally are based on safety analyses and tests which consider the predicted effects of a spectrum of postulated accidents. Such analyses should be included in Chapter 15.0. "Accident Analyses". However, having established certain functional requirements as the performance objectives of an ECCS design, this section of the SAR should include those system evaluations from which it has been concluded that functional requirements have been met with an adequate margin for contingencies. Such evaluations are expected also to provide the bases for any operational restrictions such as minimum functional capacity or testing requirements that might be appropriate for inclusion in the Technical Specifications of the license.

6.3.3.1 Results of Analyses - Analyses should be performed to demonstrate that the performance capability of the ECCS will meet the acceptance criteria of the Commission's Interim Policy Statement, issued on June 19, 1971, and any amendments thereto, using a suitable evaluation model. Describe the assumptions used and the analytical model and discuss the bases for its validity. Provide the results of these analyses. The specific information required is as follows:

For PWRs

(1) Discuss the evaluation model including reference to the evaluation model acceptable to the Commission as described in Appendix A, Parts 1, 3, 4 or 5 of the Interim Policy Statement for the appropriate nuclear steam supply system. Any deviations in the evaluation model used in the analyses from that described in the applicable Part of Appendix A of the Interim Policy Statement should be discussed in detail.

(2) For the break size range, location and type mentioned in the applicable part of Appendix A of the Interim Policy Statement, provide

the following information as a function of time: (a) the system pressure; (b) the core flow rate, pressure drop, and inlet and exit quality; (c) the flow rate out of the pipe break; (d) emergency core coolant discharge flow rate into the reactor coolant system; (e) the core reflood rate; (f) the core and downcomer liquid level during reflood; and (g) fluid temperature, heat transfer coefficient and cladding temperature at the hot spot.

(3) In evaluating breaks smaller than those analyzed using an evaluation model described in the Interim Policy Statement, the method of analysis and the results should be presented.

(4) The presentation of the evaluation results should include curves showing percent fuel rod perforations versus pipe break size analyzed.

For BWRs

(1) Discuss the evaluation model including response to the evaluation model acceptable to the Commission, as described in Appendix A, Part 2 of the Interim Policy Statement. Any deviations in the evaluation model used in the analyses from that described in Appendix A, Part 2 of the policy statement should be discussed in detail.

(2) Provide curves of peak clad temperature and percent clad metal-water reaction as a function of pipe break size for the various combinations of ECC subsystems evaluated by using the single failure criterion indicated in Table 2-1 of the topical report: "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors", NEDO-10329. A discussion should be included showing the justification for the ECC subsystem combinations used in the evaluation.

(3) For several breaks that typify small, intermediate and large breaks, provide curves of (a) peak fuel clad temperature for various rod groups, (b) core flow, (c) fuel channel inlet and outlet quality, (d) heat transfer coefficients, (e) reactor vessel pressure and water level, and (f) minimum critical heat flux ratio (MCHFR) as functions of time. Indicate the time that effective core cooling is initiated, the time the fuel channel becomes wetted based upon item 4 of Appendix A, Part 2, and the time that the temperature transient is terminated.

(4) For the analyses performed in (2) and (3) above, discuss the range of peaking factors studied and the basis for selecting the combination that resulted in the most severe thermal transient. Curves showing percent fuel rod perforations versus pipe break size analyzed, should be included.

(5) The results pertaining to the range of pipe break sizes analyzed should be summarized to permit evaluation of the extent of conformance with the Commission's Interim Acceptance Criteria delineated in the Interim Policy Statement. The system performance and core mechanical responses that may be described in other parts of the SAR should be referenced to demonstrate conformance with all four Interim Acceptance Criteria.

In addition to the above, provide the following information:

(1) Describe the results of analyses and tests performed to determine the nuclear, mechanical and chemical effects of system operation on the core.

(2) Discuss the extent to which components or portions of the ECCS are required for operation of other systems and the extent to which components or portions of other systems are required for operation of the ECCS. An analysis of how these dependent systems would function should include system priority (which system takes preference); conditions when various components or portions of one system function as part of another system, for example, when the water level in the reactor is below a limiting value, the recirculation pumps (i.e., residual or decay heat removal pumps), or feed pumps will supply water to the safety injection system and not the containment spray system; and any limitations included to assure minimum capability (e.g., storage facility common to both core cooling and containment spray systems shall have provisions whereby the quantity available for core cooling will not be less than some specified quantity).

(3) Discuss the range of acceptable lag times associated with system operation; that is, the period between the time an accident has occurred requiring the operation of the system and the time emergency core cooling flow is discharged into the core. Analysis supporting the selection should include valve opening time, pump starting time, and other pertinent parameters.

(4) Discuss thermal shock considerations, both in terms of effect on operability of the ECCS and the effect on connecting systems.

(5) State the bounds within which principal system parameters must be maintained in the interests of constant standby readiness; e.g., such things as, the minimum poison concentrations in the coolant, minimum coolant reserve in storage volumes, and minimum inoperable components.

6.3.4 Tests and Inspections

The emergency core cooling system is a standby system, not normally operating. Consequently, a measure of the readiness of the system to

operate in the event of an accident must be achieved via tests and inspections. The periodic tests and inspections planned should be identified and reasons explained as to why the program of testing planned is believe to be appropriate. The information should include such things as:

(1) What tests have been planned and why.

(2) Considerations that led to periodic testing and the selected test frequency.

(3) Test methods to be used.

(4) Requirements set for acceptability of observed performance and the bases for them.

(5) A description of the program for inservice inspection, including items to be inspected, accessibility requirements, and the types and frequency of inspection.

Evaluations made elsewhere in the SAR that explain the bases for tests planned need not be repeated but only cross-referenced.

Particular emphasis should be given to those surveillance type tests that are of such importance to safety that they may become a part of the Technical Specifications of an operating license. The bases for such surveillance requirements should be developed as a part of the SAR.

6.3.5 Instrumentation Application

This section should discuss the instrumentation provisions for various methods of actuation (e.g., automatic, manual, different locations). The conditions requiring system actuation together with the bases for the selection (e.g., during periods when the system is to be available, whenever the reactor coolant system pressure is less than some specified pressure, the core spray system will be actuated automatically) should be included in the discussion. Design details and logic of the instrumentation should be discussed in Chapter 7.0 of the SAR.

6.X Other Engineered Safety Features

The engineered safety features included in reactor plant designs vary from facility to facility. Accordingly, for each engineered safety feature, component or system provided in a facility and not already referred to in this chapter of the Standard Format, the SAR should include separate sections (numbered 6.4 through 6.X) patterned after the above and providing information on:

6.X.1 Design Bases

6.X.2 System Design

6.X.3 Design Evaluation

6.X.4 Tests and Inspections

6.X.5 Instrumentation Application

7.0 INSTRUMENTATION AND CONTROLS

The reactor instrumentation senses the various reactor parameters and transmits appropriate signals to the regulating systems during normal operation, and to the reactor trip and engineered safety feature systems during abnormal and accident conditions. The information provided in this chapter should emphasize those instruments and associated equipment which constitute the protection system (as defined in IEEE Std 279-1971 "IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations"). The discussion of regulating systems and instrumentation should be limited to considerations of regulating system-induced transients which, if not terminated in a timely manner, would result in fuel damage, radiation release, or other public hazard. Details of seismic design and testing should be provided in Section 3.10.

7.1 Introduction

7.1.1 Identification of Safety Related Systems

List all instrumentation and control systems and supporting systems that are required to function to achieve the system responses assumed in the safety evaluations, and those needed to shut down the plant safely. Also list all other systems required for the protection of the health and safety of the public.

7.1.2 Identification of Safety Criteria

List all design bases, criteria, safety guides, information guides, standards and other documents that will be implemented in the design of the systems listed in 7.1.1.

The following specific information should be included:

(1) A description should be presented of the quality assurance to be applied to the equipment in the reactor protection system, engineered safety feature circuits, and the emergency power system. This description should include the quality assurance procedures to be used during equipment fabrication, shipment, field storage, field installation, system and component checkout, and the records pertaining to each of these. Any exceptions to IEEE Std 336-1971, "IEEE Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations," should be described and justified.

(2) The criteria and their bases should be presented that establish the minimum requirements for preserving the independence of redundant reactor protection systems, engineered safety feature systems and Class IE Electric Systems* through physical arrangement and separation and for assuring the minimum required equipment availability during any design basis event.* A discussion should be included of the administrative responsibility and control to be provided to assure compliance with these criteria during the design and installation of these systems. The criteria and bases for the installation of electrical cable for these systems should, as a minimum, address:

(a) Cable derating and cable tray fill.

(b) Cable routing in congested areas and areas of hostile environment.

(c) Sharing of cable trays with non-safety related cables or with cables of the same system or other systems.

(d) Fire detection and protection in the areas where cables are installed.

(e) Cable and cable tray markings.

(f) Spacing of wiring and components in control boards, panels, and relay racks.

(3) Describe and justify any exceptions to IEEE No. 323 (April 1971), "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations."

(4) A description should be provided of the means proposed to identify physically the reactor protection system and engineered safety feature equipment as safety related equipment in the plant to assure appropriate treatment, particularly during maintenance and testing operations. The description should include the identification scheme used to distinguish between redundant channels of these systems and a discussion of how it will be evident to the operator or maintenance

*Class IE electric systems and design basis events are defined in IEEE Std. 308-1971, "IEEE Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations."

craftsman without the necessity for consulting any reference material, whether equipment, cabling, etc., is safety related and, if safety related, which channel is involved.

(5) Describe and justify any exceptions to IEEE No. 317 (April 1971), "IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations."

7.2 Reactor Trip System

For standardized systems it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.2.1 Description

Provide a description of the reactor trip system to include initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described (reference may be made to other sections of the SAR). Those parts of any system not required for safety should be identified. Provide the design basis information required by Section 3 of IEEE Std. 279-1971. Provide logic diagrams, P&I diagrams, and location layout drawings of all reactor trip systems and supporting systems. In the FSAR, provide electrical schematic diagrams for all reactor trip systems and supporting systems.

For the protection systems that actuate reactor trip, provide the following specific information:

(1) A list of those systems designed and built by the nuclear steam system supplier that are identical to those of a nuclear power plant of similar design by the same nuclear steam system supplier that has recently received a construction permit or an operating license, and a list of those that are different, with a discussion of the differences;

(2) A list of those systems and their suppliers that are designed and/or built by suppliers other than the nuclear steam system supplier; and

(3) An identification of, and justification for, those features of the design that do not conform to the criteria of IEEE Std. 279-1971, IEEE Std. 338-1971, "IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," and the AEC General Design Criteria.

7.2.2 Analysis

Provide analyses to demonstrate how the requirements of the AEC General Design Criteria, IEEE Std. 279-1971, IEEE Std. 338-1971, applicable AEC Safety Guides, and other appropriate criteria and standards are satisfied. These analyses should include, but not be limited to, considerations of instrumentation installed to prevent, or mitigate the consequences of, (a) spurious control rod withdrawals, (b) loss of plant instrument air systems, (c) loss of cooling water to vital equipment, (d) plant load rejection, and (e) turbine trip. The analyses should also discuss the need for more restrictive set points during operation with fewer than all reactor coolant loops operating. Reference may be made to other sections of the SAR for supporting systems.

7.3 Engineered Safety Feature Systems

For standardized systems it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.3.1 Description

Provide a description of the instrumentation and controls associated with the Engineered Safety Features (ESF) to include initiating circuits, logic, bypasses, interlocks, sequencing, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described (reference may be made to other sections of the SAR). Those parts of any system not required for safety should be identified. Provide the design basis information required by Section 3 of IEEE Std. 279-1971. Provide logic diagrams, P&I diagrams and location layout drawings of all ESF instrumentation and control systems and supporting systems. In the FSAR, provide electrical schematic diagrams for all ESF circuits and supporting systems.

7.3.2 Analysis

Provide analyses to demonstrate how the requirements of the AEC General Design Criteria, IEEE Std. 279-1971, IEEE Std. 338-1971, applicable AEC Safety Guides and other appropriate criteria and standards are satisfied. The method for periodic testing of engineered safety feature instrumentation and control equipment should be described. IEEE Std. 279-1971 is interpreted to require the same high degree of on-line testability for engineeered safety feature actuation as is required for the reactor trip system.

7.4 Systems Required for Safe Shutdown

For standardized systems it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.4.1 Description

Provide a description of the systems that are needed for safe shutdown of the plant, including initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described (reference may be made to other sections of the SAR). Those parts of any system not required for safety should be identified. Provide the design basis information required by Section 3 of IEEE Std. 279-1971. Provide logic diagrams, P&I diagrams and location layout drawings for these systems. In the FSAR, provide electrical schematic diagrams.

Describe the provisions taken in accordance with AEC General Design Criterion 19 to provide equipment outside the control room (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

7.4.2 Analysis

Provide analyses which demonstrate how the requirements of the AEC General Design Criteria, IEEE Std. 279-1971, applicable AEC Safety Guides and other appropriate criteria and standards are satisfied. These

analyses should include considerations of instrumentation installed to permit a safe shutdown in the event of (a) loss of plant instrument air systems, (b) loss of cooling water to vital equipment, (c) plant load rejection, and (d) turbine trip.

7.5 Safety Related Display Instrumentation

7.5.1 Description

Include a description of the instrumentation systems (including control rod position indicating systems) that provide information to the operator to enable him to perform required safety functions.

7.5.2 Analysis

Provide an analysis to demonstrate that the operator has sufficient information to perform required manual safety functions (e.g., assuring safe control rod patterns, manual engineered safety feature operations, possible unanticipated post-accident operations, and monitoring the status of safety equipment). Identify and demonstrate compliance with appropriate safety criteria.

Information should be provided to identify the information readouts or indications provided to the operator for monitoring conditions in the reactor, the reactor coolant system, and in the containment and safety-related process systems throughout all operating conditions of the plant, including anticipated operational occurrences and accident and post-accident conditions. The information should include the design criteria, the type of readout, number of channels provided, their range, accuracy and location, and a discussion of the adequacy of the design.

7.6 All Other Systems Required for Safety

This section should contain information on all other systems required for safety that are not included under Reactor Trip, Engineered Safety Features, Shutdown, Safety Related Display Instrumentation Systems or any of their supporting systems, (e.g., cold water slug interlocks, refueling interlocks and interlocks that prevent overpressurization of low pressure systems).

7.6.1 Description

Provide a description of all systems required for safety not already discussed, including initiating circuits, logic, bypasses, interlocks, redundancy, diversity, and actuated devices. Any supporting systems should be identified and described (reference may be made to other sections of the SAR). Those parts of any system not required for safety should be identified. Provide the design basis information required by Section 3 of IEEE Std. 279-1971. For an FSAR, sufficient schematic diagrams should be provided to permit an independent evaluation of compliance with the safety criteria.

7.6.2 Analysis

Provide analyses to demonstrate how the requirements of the AEC General Design Criteria, IEEE Std. 279-1971, IEEE Std. 338-1971, applicable AEC Safety Guides and other appropriate criteria and standards are satisfied. These analyses should include, but not be limited to, considerations of instrumentation installed to prevent, or mitigate the consequences of, (a) cold water slug injections, (b) refueling accidents, and (c) over-pressurization of low pressure systems. Reference may be made to other sections of the SAR for supporting systems.

7.7 Control Systems

For standardized systems it is preferred that the information listed be supplied in a topical report and that the topical report be referenced in the appropriate place in the SAR.

7.7.1 Description

Describe those control and instrumentation systems whose functions are not essential for the safety of the plant. The description should permit an understanding of the way the reactor and important subsystems are controlled.

The following information should be provided with regard to the control systems designed by the nuclear steam system supplier:

(1) Identification of the major plant control systems (e.g., primary temperature control, primary water level control, steam generator water level control) that are identical to those in a nuclear power plant of similar design by the same nuclear steam system supplier that has recently received a construction permit or an operating license; and

(2) A list and discussion of the design differences in those systems not identical to those used in the reference nuclear power plant. This discussion should include an evaluation of the safety significance of each design difference.

7.7.2 Analysis

Provide analyses to demonstrate that these systems are not required for safety. The analyses should demonstrate that the protection systems are capable of coping with all (including gross) failure modes of the control systems.

CHAPTER 8.0 ELECTRIC POWER

The electric power system is the source of power for the reactor coolant pumps and other auxiliaries during normal operation, and for the protection system and engineered safety features during abnormal and accident conditions. The information in this chapter should be directed toward establishing the functional adequacy of the emergency power sources, and assuring that these sources are redundant, independent, testable and otherwise in conformity with current criteria. Details of seismic design and testing should be provided in Section 3.10.

8.1 Introduction

A brief description of the utility grid and its interconnection to other grids should be supplied. The onsite electric system should be described briefly in general terms. Identify the safety loads, i.e., the systems and devices that require electric power to perform their safety functions. The safety functions (e.g., emergency core cooling, containment cooling, safe shutdown) and the type of electric power (a-c or d-c) should be identified for each safety load. List all design bases, criteria, safety guides, standards and other documents that will be implemented in the design of the above systems.

8.2 Offsite Power System

8.2.1 Description

Provide an analysis to demonstrate compliance with the AEC General Design Criteria (GDC), AEC Safety Guides, and other applicable standards and criteria. In particular, the two circuits required by GDC 17 to supply power for safety loads from the transmission network should be identified and shown to meet GDC 17. Describe and provide layout drawings of the circuits that connect the onsite distribution system to the preferred power supply. Include transmission lines, switchyard arrangement, rights-of-way, etc. Provide the results of the analysis that demonstrates that loss of the nuclear unit or the most critical unit on the grid will not result in loss of offsite power to the nuclear unit safety buses.

(2) Cooling System for Reactor Auxiliaries - Discuss the capability of the reactor system auxiliaries to meet the single failure criterion, the ability to withstand adverse environmental occurrences, requirements for normal operation and for operating during and subsequent to postulated accident conditions including loss of offsite power, and requirements for leakage detection and containment of leakage. Include a failure analysis to demonstrate that a single failure will not result in the loss of all, or a portion of, the cooling function (considering failures of active and passive components, and diverse sources of electric power for pumps, valves and control purposes), the means for precluding the leakage of activity to the outside environment, leakage detection provisions, prevention of long term corrosion which may degrade system performance, and safety implications related to sharing (for multiple unit facilities).

(3) Demineralized Water Make-Up System

(4) Potable and Sanitary Water Systems

(5) Ultimate Heat Sink - Describe the ultimate heat sink to be used to dissipate waste heat from the reactor facility during normal and emergency shutdown conditions. Additional guidance regarding acceptable features of ultimate heat sink facilities will be given in an AEC Safety Guide in preparation.

(6) Condensate Storage Facilities - Include discussion of the environmental design considerations, requirements for leakage control (including mitigation of environmental effects), limits for radioactivity concentration, code design requirements, and material compatibility and corrosion control. Evaluate provisions for assuring a minimum supply of condensate for emergency purposes, and provide an analysis of storage facility failure and provisions for mitigating environmental effects. The evaluation of radiological considerations should be presented in Chapter 12.

9.3 Process Auxiliaries

This section of the SAR should provide discussions of each of the auxiliary systems associated with the reactor process system. Because these auxiliary systems vary in number, type, and nomenclature for various plant designs, the Standard Format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 9.3.1 through 9.3.x) for each of the systems. These subsections should provide information on (1) design bases, (2) system description, (3) safety evaluation, (4) tests and inspections, and (5) instrumentation applications for each system.

The following paragraphs provide examples of systems that should be discussed, as appropriate to the individual plant, and identify some specific information that should be provided in addition to the items identified above.

(1) Compressed Air Systems - Describe the compressed air systems that provide station air for service and maintenance uses and include discussion of provisions for meeting the single failure criterion, air cleanliness requirements, and environmental design requirements. The evaluation of the compressed air system should include a failure analysis (including diverse sources of electric power), maintenance of air cleanliness to assure system reliability, and safety implications related to sharing (for multiple unit facilities).

(2) Process Sampling System - The design bases for the sampling system for the various plant fluids should include consideration of sample size and handling to assure that a representative sample is obtained, requirements to preclude hazards to plant personnel, and system pressure, temperature and code requirements. The points from which samples will be obtained should be delineated. The evaluation of the sampling system should provide assurance that representative samples will be obtained, and that sharing (for multiple unit facilities) will not adversely affect plant safety. The radiological evaluation for normal operation should be provided in Chapter 12.

(3) Equipment and Floor Drainage System - Describe the drainage systems for collecting the effluent from radioactive and non-radioactive drains from various specified equipment items and buildings. An evaluation of radiological considerations for normal operation, including the effects of sharing (for multiple unit facilities), should be presented in Chapters 11 and 12.

(4) Chemical and Volume Control System - The design bases for the chemical and volume control system should include consideration of the capability for the control of reactor coolant chemistry for reactivity and corrosion control, capability for maintaining the required reactor coolant system inventory, code design requirements, and environmental design conditions. The evaluation of the chemical and volume control system should include a malfunction analysis, an analysis of the capability to control the concentrations of tritium, boron, and other chemicals in the reactor coolant system, the provisions made to detect and control leakage, an analysis of the availability and reliability of the system (including heat tracing), and an analysis of the capability to isolate the system in the event of pipe breaks outside containment. The radiological evaluation for normal operation should be presented in Chapter 11 and 12.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Control Room

The design bases for the air treatment system for the control room should be provided and include ability to meet the single failure criterion, ambient temperature requirements, criteria for plant operator comfort and safety, requirements for radiation protection and monitoring of abnormal radiation levels, and environmental design requirements.

A description should be presented of the air treatment systems for the control room, including drawings.

An evaluation of the control room air treatment system should be provided and should include discussion of ability to detect air-borne contaminants (smoke, radiation, etc.) and preclude their admission to the control room or expedite their discharge from the control room, capability of filters for iodine and particulate removal, ability to meet the single failure criterion, and capability for assuring required ambient temperature level and anticipated degradation of control room equipment performance if temperature levels are exceeded. Analysis of dose levels in the control room under accident conditions should be presented in Chapter 15.

The inspection and testing requirements for the control room air treatment system should be described.

9.4.2 Auxiliary Building

A description of the heating and ventilating system for the various items of equipment in the Auxiliary Building, including drawings, should be provided. Required and design ambient temperature limits should be listed. Discuss the design bases, system design, design evaluation, test and inspection requirements and instrumentation applications.

9.4.3 Radwaste Area

The design bases for the air handling system for the radwaste area should be presented and should include requirements for meeting the single failure criterion, ambient temperature limits, preferred direction of air flow from areas of low potential radioactivity to areas of higher potential radioactivity, differential pressures to be maintained and measured, requirements for monitoring of abnormal radiation levels, and requirements for treatment of exhaust air.

A description should be provided of the air handling system for the radwaste area, including drawings.

An evaluation of the radwaste area air handling system should be presented including a system failure analysis (including effects of inability to maintain preferred air flow patterns). Evaluation of radiological considerations for normal operation should be presented in Chapters 11 and 12.

The inspection and testing requirements for the radwaste area air handling system should be provided.

9.4.4 Turbine Building

The design bases for the air handling system for the turbine-generator area in the Turbine Building should be presented and should include ambient temperature limits, preferred direction of air flow from areas of low potential radioactivity to areas of higher potential radioactivity, requirements for monitoring of abnormal radiation levels, and requirements for treatment of exhaust air.

A description should be provided of the air handling system for the Turbine Building, including drawings.

An evaluation of the Turbine Building air handling system should be presented including a system failure analysis (including effects of inability to maintain preferred air flow patterns). Radiological considerations for normal operation should be evaluated in Chapters 11 and 12.

The inspection and testing requirements for the Turbine Building air handling system should be provided.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

The design bases for the fire protection system should be provided and should include extent of station coverage, type of fire extinguishing equipment and material to be provided for each area, requirements for fire monitoring, criteria for minimizing the potential for fires, requirements to assure that operation of the fire protection system would not produce an unsafe condition, seismic design criteria for the fire protection system, and requirements to assure that failure of any portions of the fire protection system not designed to Category I requirements would not damage other Category I equipment.

A description of the fire protection and detection system, including drawings, should be provided.

An evaluation of the fire protection and detection system should be presented and should include an analysis of potential adverse effects of fire protection system operation (such as flooding of engineered safety feature equipment), design features incorporated in the unit design to minimize the potential for fire occurrences, and an analysis of the reliability of fire detection equipment.

The inspection and testing requirements for the fire protection system should be provided.

9.5.2 Communications Systems

The design bases for the communications systems for intra-plant and plant-to-offsite communications should be provided and should include requirements to meet the single failure criterion and use of diverse system types.

A description of the communication systems should be provided.

An evaluation of the communication systems should be provided and should include a failure analysis to demonstrate that the single failure criterion is met.

The inspection and testing requirements for the communication systems should be provided.

9.5.3 Lighting Systems

A description of the normal and emergency lighting system for the plant should be provided.

9.5.4 Diesel Generator Fuel Oil System

The design bases for the fuel oil system for the diesel generator should be provided and should include the requirement for onsite storage capacity, ability to meet the single failure criterion, code design requirements, and environmental design conditions.

A description of the diesel generator fuel oil system, including drawings, should be provided.

An evaluation of the fuel oil system should be provided and should include the potential for material corrosion, a failure analysis to demonstrate capability to meet the single failure criterion, ability to withstand environmental design conditions, and the planning accomplished for the procurement of additional oil, if required.

10.0 STEAM AND POWER CONVERSION SYSTEM

This chapter of the Safety Analysis Report should provide information concerning the facility steam and power conversion system. For purposes of this chapter, the steam and power conversion system (heat utilization system) should be considered to include:

(1) The steam system and turbine generator units of an indirect-cycle reactor plant, as defined by the secondary coolant system, or

(2) The steam system and turbine generator units in a direct-cycle plant, as defined by the system extending beyond the reactor coolant system isolation valves.

There will undoubtedly be many aspects of the steam portion of the facility that have little or no relationship to protection of the public against exposure to radiation. The Safety Analysis Report is, therefore, not expected to deal with this part of the facility to the same depth or detail as those features playing a more significant safety role. Enough information should be provided to allow understanding in broad terms of what the secondary plant (steam and power conversion system) is, but emphasis should be on those aspects of design and operation that do or might affect the reactor and its safety features or contribute toward the control of radioactivity. The capability of the system to function without compromising directly or indirectly the nuclear safety of the plant under both normal operating or transient situations should be shown by the information provided. Where appropriate, the evaluation of radiological aspects of normal operation of the steam and power conversion system and subsystems should be summarized in this chapter, and presented in detail in Chapters 11 and/or 12.

10.1 Summary Description

A summary description should be provided of the steam and power conversion system, indicating principal design features. An overall system flow diagram and a summary table of the important design and performance characteristics should be included. The description should indicate the system design features that are safety related.

10.2 Turbine-Generator

The design bases for the turbine-generator equipment should be provided and should include the performance requirements under both normal operating and transient conditions, intended mode of operation (base loaded or load following), functional limitations imposed by the design or operational characteristics of the reactor coolant system (rate at which electrical load may be increased or decreased with and without reactor control rod motion or steam bypass), and design codes to be applied.

A description of the turbine-generator equipment including moisture separation, use of extraction steam for feedwater heating, and control functions which could influence operation of the reactor coolant system, should be provided including drawings.

An evaluation of the turbine-generator and related steam handling equipment should be provided. This evaluation should include a summary discussion of the anticipated operating concentrations of radioactive contaminants in the system, reduction levels associated with the turbine components and resulting shielding requirements, and the extent of access control necessary based on radiation levels and shielding provided. Details of the radiological evaluation should be provided in Chapters 11 and 12.

10.3 Main Steam Supply System

The design bases for the main steam line piping from the steam generator in the case of an indirect cycle plant, or from the outboard isolation valve in the case of a direct cycle plant, should be provided and should include performance requirements, environmental design criteria, inservice inspection requirements, and design codes to be applied.

A description should be provided of the main steam line piping including drawings showing interconnected piping.

An evaluation of the design of the main steam line piping should be provided and should include an analysis of the ability to withstand limiting environmental conditions, and provisions for permitting in-service inspections to be performed.

The inspection and testing requirements of the main steam line piping should be described. Describe the proposed requirements for preoperational and inservice inspection of steam-line isolation valves, or cross-reference other sections of the SAR where this is described.

10.4 Other Features of Steam and Power Conversion System

This section of the SAR should provide discussions of each of the principal design features and subsystems of the steam and power conversion system. Because these systems vary in number, type, and nomenclature for various plant designs, the Standard Format does not assign specific subsection numbers to these systems. The applicant should provide separate subsections (numbered 10.4.1 through 10.4.x) for each. These subsections should provide information on (1) design bases, (2) system description, (3) safety evaluation, (4) tests and inspections, and (5) instrumentation applications for each subsystem or feature.

The following paragraphs provide examples of subsystems and features that should be discussed, as appropriate to the individual plant, and identify some specific information that should be provided in addition to the items identified above.

(1) Main Condensers - The description of the main condensers should include performance requirements, anticipated inventory of radioactive contaminants during normal operation and during shutdown, anticipated air leakage limits, control functions which could influence operation of the reactor coolant system, and potential for hydrogen build-up.

(2) Main Condensers Evacuation System - The description of the evacuation systems for the main condensers should include performance requirements for start-up and normal operation, anticipated radioactive contamination discharge rates, evaluation of the capability to limit or control loss of radioactivity to the environment, and control functions which could influence operation of the reactor coolant system. Details of the radiological evaluation should be provided in Chapter 11.

(3) Turbine Gland Sealing System - The discussion of the turbine gland sealing system should include identification of the source of non-contaminated steam, a failure analysis to provide an estimate of potential radioactivity leakage to the environment in the event of a malfunction, and discussion of the means to be used to monitor system performance. The inspection and testing requirements should be described. Details of the radiological evaluation should be provided in Chapter 11.

(4) Turbine Bypass System - The design bases for the turbine bypass system should include performance requirements, requirements for meeting the single failure criterion, design codes to be applied, and environmental design criteria. The evaluation of the turbine bypass system should include a failure analysis to determine the effect of equipment malfunctions on the reactor coolant system, and an analysis to assess the ability to withstand environmental phenomena.

(5) Circulating Water System - The description of the circulating water system should include discussion of performance requirements, dependence upon the system for emergency cooling, control of the circulating water chemistry, and potential physical interaction of cooling towers, if any, with the plant structure.

(6) Condensate Clean-up System - The design bases for the condensate clean-up system should include the fraction of condensate flow to be treated, impurity levels to be maintained, and design codes to be applied. The evaluation of the condensate clean-up system should include an analysis of anticipated impurity levels, an analysis of the contribution of impurity levels from the secondary system to reactor coolant system activity levels, and performance monitoring.

(7) Condensate and Feedwater Systems - The design bases for the condensate and feedwater systems should include design codes to be applied, criteria for isolation from the steam generator or reactor coolant system, inservice inspection requirements, and environmental design requirements. The evaluation of the condensate and feedwater systems should include an analysis of component failure, effects of equipment malfunction on the reactor coolant system, and an analysis of isolation provisions to preclude release of radioactivity to the environment in the event of a pipe break.

(8) Steam Generator Blowdown Systems - The design bases for the steam generator blowdown system should include performance requirements, sampling criteria, isolation criteria, design codes to be applied, environmental design criteria, and primary-to-secondary leakage limitations. The evaluation of the steam generator blowdown system should include an analysis of radioactivity discharge rates, a failure analysis of system components, system performance during abnormally high primary-to-secondary leakage, and an analysis of steam generator shell-side radioactivity concentration during system isolation. Details of the radiological evaluation for normal operation should be presented in Chapters 11 and 12.

The inspection and testing requirements for the steam generator blowdown system should be provided.

11.0 RADIOACTIVE WASTE MANAGEMENT

The purpose of the information to be provided in this chapter is to provide assurance that the nuclear plant has sufficient installed capacity and treatment equipment in the radioactive waste (radwaste) systems to reduce the radioactivity to levels which will not be in excess of the appropriate limits for the general public or plant personnel and are as low as practicable. Wherever appropriate, summary tables should be provided.

11.1 Source Terms

The sources of radioactivity which serve as input into the various radioactive waste systems should be defined explicitly. The mathematical model used to determine the specific activity of each isotope in the primary coolant should be given and all assumptions justified. In addition to a presentation of the specific isotopic inventory in the coolant, the isotopic inventory in the fuel plenums and gaps for the entire core should also be presented. The delineation of all the activities in the coolant and in the plenum and gap of the fuel elements should, at a minimum, take into account the power densities of the core, burnups and fuel failure which are consistent with experience and design. State the fraction of plenum and gap activity assumed to be released to the coolant. The fraction which is chosen should be consistent with past experience, heat loadings on the fuel pins and stresses caused by anticipated operational occurrences. Discuss the fuel experience that has been gained for the type of fuel that will be used, including the failure experience, the burnup experience, and the thermal conditions under which the experience was gained. If this information is presented in other sections of the SAR, only cross-referencing is necessary.

If escape rate coefficients are used, a justification of each number used should be presented. The variation of the escape rate coefficients with power densities and half-life should be presented and justified. The basis upon which each escape rate coefficient is derived should be presented.

A complete derivation and justification of activated corrosion source terms should be presented. All assumptions used in the derivation should be stated. The activation of water and constituents ordinarily found in the makeup to the reactor coolant system should also be taken into account. Production of isotopes (e.g., N-16) should be listed and justified. Previous pertinent experience should be cited.

In order to evaluate the adequacy of various ventilation systems, provide estimates of the leakage rate from the reactor coolant system and other fluid systems containing radioactivity. Summarize the sources of leakage and estimate their contribution to the total quantity. Provide estimates of the escape of gases from each leakage source and describe their subsequent transport and release. State and justify all assumptions. Cite previous pertinent experience. Discuss leakage measurements and control methods. The principal discussions of coolant leakage in other sections of the SAR should be cross-referenced.

11.2 Liquid Waste Systems

11.2.1 Design Objectives

The design objectives of the various liquid waste systems should be stated in terms of expected annual activity releases (by nuclide), and exposures to individuals and the population in light of the requirements of 10 CFR Parts 20 and 50.

11.2.2 Systems Descriptions

The input waste streams into the various subsystems of the radioactive liquid waste system should be identified by concentration by nuclide and flow rate on process flow diagrams. Concentrations and quantities for both normal operation and for conditions resulting from anticipated operational occurrences should be provided. The source term of radioactivity for each input stream should be identified and justified. Detailed process flow diagrams should be presented; the principal flow paths through each system should be indicated clearly (for example, by use of multi-colored process lines). Identify vents, drains, and secondary flow paths for each system. Indicate the effect of each process on the streams. All bypasses through which waste could circumvent process equipment and be released to the environment and all discharge points to the environment should be indicated clearly. To provide information for use in the evaluations of Chapter 12, those lines containing significant radioactivity that are to be field-run should be indicated on the process flow diagrams. All systems that are used to reduce levels of radioactivity in liquid effluents should be included. State the capacity and expected decontamination factor for each isotope for each piece of equipment. Cite pertinent previous experience.

11.2.3 Operating Procedures

The operating procedures that will be used for all liquid radwaste management equipment should be described. Cite pertinent previous experience on the effectiveness of such procedures.

11.2.4 Performance Tests

Performance tests that will be used on a periodic basis to verify the decontamination factors and other aspects of a given design should be stated. Cite pertinent previous experience with such tests.

11.2.5 Estimated Releases

The expected releases from the liquid radwaste system in curies per year per nuclide should be stated separately for each liquid system. The expected releases should cover normal operation, and anticipated operational occurrences. Relate the expected releases to the Technical Specifications proposed for gaseous effluents.

11.2.6 Release Points

All release points from the liquid radwaste systems to the environment should be identified clearly on process flow diagrams, on general arrangement drawings and on a site plot plan.

11.2.7 Dilution Factors

All dilution factors that are used in evaluating the release of radioactive effluents should be stated and justified. Recirculation of effluents from discharge to intakes should be considered.

11.2.8 Estimated Doses

Based on the information given in the above, estimate the following doses that would be received by the general public as a result of releasing the radioactive effluents by the paths and with the dilution factors mentioned above:

a. The maximum whole body dose to an individual (rem);

b. The maximum organ dose to an individual (rem);

c. The whole body dose to the population (man-rem).

11.3 Gaseous Waste Systems

11.3.1 Design Objectives

The design objectives of the various gaseous waste systems should be stated in terms of expected annual activity releases (by nuclide) and exposures to individuals and the population, in the light of the requirements of 10 CFR Parts 20 and 50. As used in this section, gaseous waste includes noble gases and airborne halogens and particulates.

11.3.2 Systems Descriptions

The input waste streams into the various subsystems of the radioactive gaseous waste system should be identified by concentration (by nuclide) and flow rate on process flow diagrams. The source term of radioactivity for each input should be presented; the principal flow paths through each system should be indicated clearly (for example, by use of multi-colored process lines). Identification of vents and secondary flow paths for each system should be indicated. All bypasses through which waste could circumvent process equipment and be released to the environment and all discharge points to the environment should be indicated clearly. All ducting and piping containing significant radioactivity that is to be field run should be indicated on the process flow diagrams. All systems used to reduce levels of radioactivity in gaseous effluents should be included. State the capacity and decontamination factor for each isotope for each piece of equipment. Cite pertinent previous experience.

11.3.3 Operating Procedures

The operating procedures to be used for gaseous waste systems should be described. Cite pertinent previous experience on the effectiveness of such procedures.

11.3.4 Performance Tests

Performance tests that will be used on a periodic basis to verify the decontamination factors and other aspects of a given design should be stated. Cite pertinent previous experience with such tests.

11.3.5 Estimated Releases

The expected releases from the gaseous waste systems in curies per year per nuclide should be stated separately for each system. The expected releases should cover normal operation and anticipated operational occurrences. Relate the expected releases to the Technical Specifications proposed for liquid effluents.

11.3.6 Release Points

All release points from the gaseous waste systems to the environment should be identified clearly on process flow diagrams, on general arrangement drawings, and on a site plot plan.

11.3.7 Dilution Factors

All dilution factors which are used in evaluating the release of gaseous radioactive effluents should be stated and justified.

11.3.8 Estimated Doses

Based on the information given above, estimate the following doses that would be received by the general public as a result of releasing the radioactive effluents by the paths and with the dilution factors mentioned above:

a. The maximum whole body dose to an individual;

b. The maximum organ dose to an individual from halogens and particulates;

c. The whole body dose to the population.

11.4 Process and Effluent Radiological Monitoring Systems

A complete description should be given for liquid and gaseous systems separately. Provide summary tables as appropriate. (See AEC Safety Guide No. 21.)

11.4.1 Design Objectives

State the design objectives of the radiological monitoring systems for normal operation and anticipated operational occurrences in relation to the requirements of 10 CFR Parts 20 and 50 and AEC General Design Criterion 64. Distinguish the differences between the design objectives for these situations and those for accident situations.

11.4.2 Continuous Monitoring

For each location subject to continuous monitoring provide: (a) the basis for selecting the location; (b) the expected concentrations or radiation levels; (c) the quantity to be measured (e.g., external radiation level, gross concentration, isotopic concentration); (d) the detector type, sensitivity and range, considering items (a), (b) and (c) above, and, for remote devices, the type and arrangement of the sampler and estimates of sampling line interferences or losses; (e) the type and locations of power sources and recording and indicating devices; (f) setpoints and the bases for their selection; and (g) the type and locations of annunciators and alarms, and the system or operator actions which they initiate.

11.4.3 Sampling

For each location subject to periodic sampling, provide: (a) the basis for selecting the location; (b) expected composition and concentrations; (c) the quantity to be measured (e.g., gross or isotopic concentrations; (d) sampling frequency and procedures; (e) analytical procedure and sensitivity; and (f) influence of results on plant operations.

11.4.4 Calibration and Maintenance

For every instrument or logical grouping of instruments, as appropriate, describe the procedures governing calibration and maintenance. Also describe the arrangements for obtaining independent audits and verifications.

11.5 Solid Waste System

This section should describe in detail the solid radwaste capabilities of the plant.

11.5.1 Design Objectives

The design objectives of the solid radwaste system should be stated in terms of volumes, forms and activities, and the radiation levels that can be accommodated.

11.5.2 System Inputs

The assumed system inputs based on volume or weight and isotopic curie inventories should be derived and justified. The inventories should be consistent with source terms presented under Section 11.1. Liquid and solid input streams should be identified on a detailed process flow diagram. A detailed process flow diagram for the total solid radwaste system should be presented.

11.5.3 Equipment Description

A description should be presented of all the equipment in the solid radwaste system. Capacities, through-put rates and storage capabilities should be stated. The operating procedures which will be followed in the utilization of the solid radwaste equipment should be stated. Cite pertinent previous experience with such equipment.

11.5.4 Expected Volumes

The expected volumes of solid wastes, the associated curie content and the principal nuclides that will be shipped from the site should be derived and justified. Experience from similar plants already operating should be presented.

11.5.5 Packaging

The packaging containers of the solid radwastes should be defined in detail including the type of container, the manner in which it is to be packed and the permissible levels of activity. Indicate conformance with applicable standards.

11.5.6 Storage Facilities

A detailed description should be presented of the storage facilities available for packaged solid radwastes including capacity, exact location on a plot plan and general arrangement and details for removal of the solid radwastes. State the expected onsite storage period and the decay realized by such storage.

11.5.7 Shipment

The manner in which the radwastes will be shipped from the site should be stated. The allowed locations on the site where the shipping containers or vehicles may be stored should be identified.

11.6 Offsite Radiological Monitoring Program

Describe the monitoring program with respect to its capability to determine, in conjunction with effluent monitoring, estimates of individual and population exposure beyond the site boundary, at the design levels of radiation and radioactive effluents. Wherever appropriate, differences between the preoperational and operational programs should be delineated.

11.6.1 Expected Background

Enumerate the expected (or measured) background levels of radiation and radioactivity (and their variation in time and space), both from natural and man-made sources.

11.6.2 Critical Pathways

Based on the expected liquid and gaseous releases (provided elsewhere in this chapter), describe the pathways of human exposure from plant operation likely to account for most of the exposure. Provide the mathematical models to be used to make exposure estimates, given effluent and environmental monitoring data. List and justify all assumptions made, or relevant information to be developed (e.g., reconcentration factors, food consumption rates).

11.6.3 Sampling Media, Locations and Frequency

Provide the basis for the choice of sampling media, sampling locations, and frequency of sampling in the light of 11.6.1 and 11.6.2. (The complete program need not be presented here, but must appear in the appropriate section of the Technical Specifications.)

11.6.4 Analytical Sensitivity

Describe the size and physical characteristics of each type of sample, the kinds of radiological analyses to be performed and the measuring equipment to be used, and derive and justify the sample detection sensitivity.

11.6.5 Data Analysis and Presentation

Describe the kinds of mathematical and statistical analyses to be performed on the resultant data, and give an indication of the type of format to be used in the presentation of results.

11.6.6 Program Statistical Sensitivity

Derive and justify, in the light of the parameters described above, the overall statistical sensitivity of the program to achieve its objectives of estimating probable exposures to man.

12.0 RADIATION PROTECTION

The purpose of the information to be provided in this chapter is to permit a determination that direct radiation exposures to persons at the site boundary from sources contained within the plant and on the site, and external and internal exposures to plant personnel will be kept as low as practicable and within applicable limits.

12.1 Shielding

12.1.1 Design Objectives

Describe the design objectives of plant shielding for normal operation, including anticipated operational occurrences, with respect to meeting the requirements of 10 CFR Parts 20 and 50. The maximum and average external dose rates from normal operation, including anticipated operational occurrences, that will be allowed at the site boundary and in areas within the plant where plant personnel, construction workers or site visitors are permitted should each be identified.

12.1.2 Design Description

Provide scaled layouts and cross sections of buildings that contain process equipment for treatment of radioactive fluids. Also, provide a detailed plot plan showing the total plant layout within the site boundary, and explicitly identifying all outside storage areas and the location of railroad spurs or sidings.

Describe design criteria for the erection and dimensions of shield walls, for penetrations through shield walls, and for acceptable radiation levels at valve stations for process equipment containing radioactive fluids.

To permit evaluation of the capability of the control room to meet AEC General Design Criterion 19, a layout drawing of the control room should be provided. Scaled isometric views of the control room and descriptions of all shielding required to maintain habitability of the control room during the course of accidents should be provided.

12.1.3 Source Terms

The total quantity of the principal nuclides in process equipment that contains or transports radioactivity should be identified as a function of operating history. Expected maximum and average values of the radioisotopic inventory should be stated. The sources should be consistent with those presented in Chapter 11. Provide an estimate of dose rate at the site boundary per curie of waste stored outside of buildings (including shipping casks).

Other radioactive items that are not clearly assignable to the above categories should be listed in this section and evaluated similarly. For instance, the nitrogen-16 contribution to exposure from the turbine building should be considered here.

Identify the steps taken to assure that field run process piping, that is designated to carry radioactive materials, is routed with appropriate regard for minimizing radiation exposures to plant personnel.

12.1.4 Area Monitoring

Provide the locations and specifications of the types of instruments to be used for area radiation monitoring, and the criteria used to determine the necessity for and location of the equipment. Describe their operational characteristics, including type of detector, sensitivity, range, method of calibration, setpoints (and their bases), and the location and type of annunciators and alarms (and the system or operator actions they initiate), and describe the maintenance and calibration programs to be followed. Provide the type and location of power sources, and indicating and recording devices. Indicate the manner in which data will be recorded.

12.1.5 Operating Procedures

Describe the operating procedures to assure that external exposures will be kept as low as practicable during plant operation and maintenance. Cite relevant previous experience on the effectiveness of such procedures.

12.1.6 Estimates of Exposure

Provide a summary of the estimated peak external dose rates and annual doses at selected in-plant locations, at the site boundary, at visitor centers and in the control room, from normal operation including anticipated operational occurrences. Provide an estimate of the yearly man-rem exposures from the plant as designed. Compare the estimated doses with experience from relevant operating plants. Provide justification for the thickness of shielding provided; include the geometric and physical models and basic assumptions and data employed.

12.2 Ventilation

12.2.1 Design Objectives

Describe the design objectives of the plant ventilation systems for normal operation, including anticipated operational occurrences, with respect to meeting the requirements of 10 CFR Parts 20, 50, and 100.

The maximum and average airborne radioactivity levels for normal operation, including anticipated operational occurrences, that will be allowed in areas within plant structures and on the plant site where plant personnel, construction workers, or site visitors are permitted should each be identified.

12.2.2 Design Description

Provide as complete a description as possible of the ventilation system for each building which can be expected to contain radioactive materials. The description should include building volumes, expected flow rates, and filter characteristics, and the design criteria on which these are based.

Provide a separate description of the control room ventilation system to permit evaluation of its capability to meet AEC General Design Criterion 19 with respect to inhalation dose. Indicate the locations of air intakes and describe filter characteristics.

12.2.3 Source Terms

In addition to the information provided in Section 12.1.3, also provide estimates of equipment leakage resulting in airborne radioactivity within plant buildings.

12.2.4 Airborne Radioactivity Monitoring

Provide the locations and specifications of the types of fixed instruments to be used for airborne radioactivity monitoring, and the criteria used to determine the necessity for and location of the equipment. Describe their operational characteristics, including sampling lines (if any), detector type, sensitivity, range, and calibration; filter characteristics; type and location of power sources and indicating and recording devices; setpoints and their bases; type and location of annunciators and alarms and the system or operator actions they initiate; and the maintenance and calibration programs to be followed. Describe any special portable instrument or grab sample methods used to check the fixed system. Indicate the manner in which data will be recorded.

12.2.5 Operating Procedures

Provide a description of plant operating procedures to assure that onsite inhalation exposures will be kept as low as practicable during plant operation and maintenance. Cite relevant previous experience on the effectiveness of such procedures.

12.2.6 Estimates of Inhalation Doses

The expected annual inhalation doses to plant personnel and peak air concentrations should be estimated for each building in the reactor facility. The estimates should be compared with experience from relevant operating plants. Describe the methods used and list and justify all assumptions.

12.3 Health Physics Program

12.3.1 Program Objectives

Describe the health physics program organization and objectives.

12.3.2 Facilities and Equipment

Describe the available health physics facilities and equipment, including handling methods and special shielding for external protection; respiratory equipment and protective clothing; and portable and laboratory equipment (operational characteristics, sensitivities, calibration and maintenance procedures, and their locations).

12.3.3 Personnel Dosimetry

Describe the methods and procedures for external and internal dosimetry of plant personnel, including sensitivity, calibration, processing and recording.

13.0 CONDUCT OF OPERATIONS

This chapter of the Safety Analysis Report should provide information relating to the framework within which operation of the facility will be conducted.

The operation of the facility entails a myriad of instructions and procedures of varying detail for the operating staff. The details of such procedures should not be included in the Safety Analysis Report, but information should be provided to indicate generally how the applicant intends to conduct operations, and to assure that the licensee will maintain a technically competent and safety-oriented staff.

The following sections indicate the kinds of information needed.

13.1 Organizational Structure of Applicant

13.1.1 Corporate Organization

This section should describe the structure and qualifications of the applicant's and his contractors' corporate organizations. The following specific information should be included:

(1) Corporate functions, responsibilities and authorities with respect to nuclear plant design, construction, quality assurance, testing and other applicable activities should be described.

(2) A description should be provided of the applicant's corporate management and technical support staffing and in-house organizational relationships established for the design and construction review and quality assurance functions, and of the responsibilities and authorities of personnel and organizations described in (1) above.

(3) The working interrelationships and organizational interfaces among the applicant, the nuclear steam supply system manufacturer, the architect-engineer, and other suppliers and contractors should be described.

(4) A description should be included of the applicant's corporate (home office) technical staff specifically supporting the operation of the nuclear plant, including a description of the duties, responsibilities, and authority of the "Engineer in Charge" and the assigned engineering technical staff; numbers of personnel, qualifications, educational backgrounds (disciplines) and technical experience. Technical support to the

corporate technical staff may be provided by the use of outside consultants. If such arrangements are to be used, the specific areas of responsibility and functional working arrangements of these support groups should be provided.

13.1.2 Operating Organization

This section should describe the structure, functions and responsibilities of the operating organization. The following specific information should be included:

(1) Provide a comprehensive description of the facility organizational arrangement (organization chart) to show the title of each position in the operations, technical and maintenance groupings, the number of persons assigned to common or duplicate positions (technicians, shift operators, repairmen) and the positions requiring licenses in accordance with 10 CFR Part 55. Additional guidance is provided in an AEC report, WASH-1130 "Utility Staffing for Nuclear Power."

(2) The functions, responsibilities and authorities of all personnel positions should be described, including a specific succession to responsibility for overall operation of the facility in the event of absences, incapacitation of personnel or other emergencies.

(3) Describe the proposed shift crew composition including position titles, license qualifications and number of personnel on each shift (the number of reactors and generating units, and the facility and control room layout bear a relationship to shift crew composition).

13.1.3 Qualification Requirements for Nuclear Facility Personnel

This subsection should describe the proposed minimum qualification requirements for onsite plant personnel. It is expected that these qualification requirements will meet or exceed the minimum qualification requirements set forth in the current ANSI N-18.1 document, "Standard for Selection and Training of Personnel for Nuclear Power Plants." If this is not the case, justification should be provided.

The following specific information should be included:

(1) The minimum qualification requirements should be stated for all plant operating, technical and maintenance support personnel (Plant Superintendent/Plant Manager or equivalent down through licensed and non-licensed plant operators, technicians and repairmen).

(2) The qualifications of the initial appointees to (or incumbents of) these positions should be presented in resume format for all plant managerial and supervisory technical personnel (operating, technical and maintenance). The resumes should identify individuals by name and, as a minimum, should describe the formal education, the training, and the experience (including prior AEC licensing) of the individuals.

13.2 Training Program

13.2.1 Program Description

A description of the proposed nuclear training program should be provided in the PSAR. The FSAR should provide a description of the training program as it was actually carried out, noting any changes from that described in the PSAR. Guidance on the required training is available in ANSI N-18.1 and the AEC Licensing Guide, "Operating Licenses, Division of Reactor Licensing, November 1965." The following specific information should be included:

(1) The program description should include the proposed subject matter content of the formal nuclear training program (related technical training) for Licensed Senior Reactor Operator (SRO) and Licensed Reactor Operator (RO) candidates; and the length of time to be devoted to each aspect of the training program.

(2) A chart should be provided to show the schedule of each part of the training program for each employee in relation to schedule for preoperational testing and fuel loading.

(3) Practical (on-the-job) reactor plant operation to be included as a part of the nuclear training program for RO and SRO candidates should be described, with the length of time to be devoted to this aspect of the training program.

(4) Reactor plant simulator training to be included as a part of the nuclear training program for RO and SRO candidates (if applicable) should be described with the length of time devoted to such training.

(5) Any previous nuclear training allowable to RO and SRO candidates, such as U. S. Navy Nuclear Power Training Program or other experience that may establish eligibility for RO or SRO license examination, should be described.

(6) Other formal (on-the-job) training programs to be provided for RO and SRO candidates (e. g., preoperational testing, startup,) should be described.

(7) Training programs to be provided for personnel not requiring licenses (certain managers, supervisors, professionals, operators, technicians and repairmen) should be described.

(8) General employee training to be provided to all persons regularly employed in the nuclear power plant should be described.

(9) State the position title of the individual responsible for conduct and administration of the nuclear power plant training program.

13.2.2 Retraining Program

A description of the retraining program should be provided in the FSAR, and should include the applicable items in 13.2.1 and the frequency with which retraining is accomplished.

13.2.3 Replacement Training

A description of the replacement training program should be provided in the FSAR and should include the items in 13.2.1 above.

13.2.4 Records

Both the PSAR and the FSAR should describe provisions for maintaining records of qualifications, experience, training and retraining for each member of the plant organization. The documents should also describe the methods to be used for evaluating training program effectiveness.

13.3 Emergency Planning

This section of the SAR should describe the applicant's plans for coping with emergencies. The information to be included in the PSAR is described in Section 50.34 (a)(10) of 10 CFR Part 50. The minimum items to be discussed in the preliminary plans are set forth in 10 CFR Part 50, Appendix E - Emergency Plans for Production and Utilization Facilities - Section II.

The information to be included in the FSAR is described in Section 50.34 (b)(6)(v) of 10 CFR Part 50. The minimum items to be discussed in the FSAR, which should include the final Emergency Plan, are set forth in

10 CFR Part 50, Appendix E, Section IV. Guidance on emergency planning is available in "Guide to the Preparation of Emergency Plans for Production and Utilization Facilities;" December 1970, U. S. Atomic Energy Commission.

13.4 Review and Audit

This section should describe the applicant's means for performing independent review and audit of nuclear facility operations in order to determine if the facility is being operated safely and within the terms of the license. This is usually performed by the committee method. Many applicants have an additional "plant operating review committee" made up of members of the operating staff. This should not be confused with the independent review and audit committee. Guidance on the essential elements of a satisfactorily comprehensive review and audit program is available in the proposed Standard ANS-3.2 "Standard for Administrative Controls for Nuclear Power Plants," Draft No. 6, September 1971.

13.4.1 Review and Audit - Construction

Many applicants propose to use the review and audit committee during the design and construction of the facility as part of the quality assurance program. In such a case, the applicant's PSAR should include a written charter for the review and audit group describing the group's responsibilities and administrative procedures.

The charter should include the subjects within the purview of the group, and the mechanism for convening meetings (if not always periodic). The charter should indicate the provisions for the use of subgroups and consultants. The responsibility for appointment of members of the group should be indicated and the time (in relation to scheduled fuel loading) that the group will be appointed and functional should be stated. Describe the responsibility and authority of the group and the requirements for recording, reporting, approval and dissemination of meeting minutes and other reports of its activities.

13.4.2 Review and Audit - Test and Operation

In the FSAR, the information indicated in 13.4.1 should be provided and the following additional information added:

(1) The composition of the group (numbers and qualifications) established to audit and evaluate both personnel and equipment should be stated. The measures to prevent degradation of the qualifications of the review and audit groups should be described, including alternate members who may serve in lieu of regular members (usually in the form of minimum qualifications requirements for various technical specialties or disciplines associated with nuclear power facilities). Where outside consultants are used on the review and audit group, qualifications and active participation, including voting rights, should be delineated.

(2) The meeting frequency should be stated.

(3) The quorum required to conduct business and designation of non-voting members, if any, should be stated.

13.5 Plant Procedures

This section of the SAR should include a commitment to conduct safety-related operations by detailed written procedures. The FSAR should include a list of titles of procedures (that indicate clearly their purpose and applicability), and a description of the review, change and approval procedures for all plant operating, maintenance, and testing procedures.

Plant procedures should be in accordance with guidance contained in Proposed Standard ANS 3.2.

13.6 Plant Records

This section of the SAR should include a commitment to keep a recorded history of the facility, in accordance with 10 CFR Part 50, Appendix B, Section XVII, Quality Assurance Records. Further guidance regarding maintenance of plant records is provided in the proposed standard ANS 3.2.

The FSAR should describe provisions for maintaining operating records such as power levels, of principal maintenance activities and of abnormal occurrences for specified periods of time (usually 5 to 6 years).

Provisions should be described in the FSAR for maintaining records of occurrences such as radioactive releases and environmental surveys, which are generally kept for the service life of the facility.

13.7 Industrial Security

This section should describe the applicant's plans for protection against industrial sabotage, in accordance with guidance contained in AEC Safety Guide 17, "Protection Against Industrial Sabotage." Further guidance is available in proposed Standard ANS 3.2. Detailed security measures for the physical protection of the facility against industrial sabotage may be withheld from public disclosure as provided in Section 2.790 of 10 CFR Part 2.

13.7.1 Personnel and Plant Design

This subsection should describe the organization, administration, and conduct of the industrial security program. Describe those features of the plant design and arrangement that enhance industrial security and reduce the vulnerability of the plant to deliberate acts which may adversely affect the plant and public safety.

Describe personnel selection policies, employee performance and evaluation procedures, and the industrial security training program used to assure that reliable and emotionally stable personnel are selected, maintained, and assigned to the plant staff.

13.7.2 Security Plan

The FSAR should include the following additional information:

(1) Means for control of access should be described, including administrative and physical personnel and material controls, such as: security measures to be employed at the exclusion area radius or site boundary, entrances to the reactor control room, building, containments, vital equipment areas and rooms where intentional or unintentional manipulations of controls or other actions would seriously affect plant operations and safety; alarm and electrical/electronic protection and surveillance systems or devices; provisions for the manning and operation of access control points.

(2) Measures should be described for the control of personnel by categories; general visitors, utility employees not members of the regular plant staff, contractor and vendor personnel, and plant staff, including personnel monitoring and accountability controls.

(3) Describe the general methods of controlling access in emergencies such as fires and industrial accidents (i.e., compatibility of industrial security plan with emergency plans and procedures.

(4) Describe the program for surveillance and monitoring of vital equipment, components and sensitive materials such as nuclear fuel and radioactive sources. The description should include the methods established for detecting physical changes in the status of equipment, components or materials on a periodic basis, such as the operational availability of engineered safeguards, valve positions and inspection of nuclear fuel upon receipt.

(5) Discuss measures for dealing with potential security threats and the liaison developed with Federal, state and local law enforcement agencies. This section should include a statement that incidents involving attempted or actual breach of industrial security controls or attempted acts of sabotage, will be reported to the Commission within 24 hours.

(6) Describe administrative procedures developed for investigation of security incidents, reports and audits of the industrial security program.

(3) Describe the test objectives and the general methods for accomplishing these objectives, the acceptance criteria that will be used to evaluate the test results, the general prerequisites for performing the tests, including special conditions to simulate normal and abnormal operating conditions of the tests listed.

(4) Discuss the procedures that will guide fuel loading, attainment of initial criticality, and ascension to power, including safety and precautionary measures to be used to assure safe operation of the plant.

(5) Describe the applicant's system for checking out normal and emergency operating procedures.

14.2 Augmentation of Applicant's Staff for Initial Tests and Operation

This section of the FSAR should describe the applicant's plans for the assignment of additional personnel to supplement his staff during startup and power testing. Guidance on the required staff expansion is contained in "Guide to the Planning of Preoperational Testing Programs, USAEC, December 1970" and in "Guide for the Planning of Initial Startup Programs, USAEC, December 1970." The following specific information should be provided.

(1) The functions, responsibilities and authorities of the various organizations established as augmenting organizations to the applicant's normal operating organization during initial tests and operations should be described. The central authority of the applicant over initial tests and operations should be clearly indicated.

(2) The working interrelationships and organizational interfaces of all augmenting groups during the initial tests and operations should be specified.

(3) The functions, responsibilities and authorities of key augmenting personnel positions should be described.

(4) The qualifications of the appointees to the positions above should be presented, preferably in the form of resumes.

14.0 INITIAL TESTS AND OPERATION

This chapter of the Safety Analysis Report should provide information relating to the period of initial operation, with particular emphasis on tests planned to demonstrate the degree to which the facility does, in fact, meet the design criteria. Explanations for any special limits, conditions, surveillance requirements, and procedures to be in force during the initial period of operation and until such time as acceptable design performance is demonstrated should be included.

Throughout other parts of the SAR, limits, conditions, surveillance requirements, and procedures for facility operation may have been established. For some facilities, however, these may be made more restrictive during the period of initial operation and relaxed to their final condition only as actual operation demonstrates their acceptance from a safety viewpoint. Such matters should be discussed in this chapter.

14.1 Test Program

This section of the PSAR should include a discussion of the preoperational testing program including its objectives, a list of test titles, and a schedule of test sequence, in accordance with the guidance contained in "Guide for the Planning of Preoperational Testing Programs, USAEC, December 7, 1970." The section should also include a discussion of initial fuel loading and the startup and power ascension program including a list of tests and a schedule of test sequence, in accordance with guidance contained in "Guide for the Planning of Initial Startup Programs, USAEC December 7, 1970 (revised)." Further guidance on testing is contained in proposed Standard ANS 3.2, and in 10 CFR 50, Appendix B, Criterion XI; Subject: Test Control.

In the FSAR, the following specific information should also be included:

(1) Describe the system used for preparing, reviewing, approving and executing all testing procedures and for evaluating, documenting and approving the test results, including the organizational responsibilities and personnel qualifications for the applicant and his contractors.

(2) The administrative procedures should be described for incorporating any needed system modifications or procedure changes, based on the results of the tests (e.g., test procedure inadequacies or test results contrary to expected test results).

actions by the reactor protection system or control systems), and (4) do not lead to significant radiation exposures off site. By definition, Class 1 events do not propagate to cause a more serious event (i.e., a Class 2 or 3 event).

Class 2 events are categorized as those which result from off-design operational transients or accidents and which (1) may induce fuel failures in excess of those expected in routine operation (i.e., from fuel cladding defects), (2) may lead to a breach of barriers to fission product release or of primary system boundary, (3) may require operation of engineered safety features (including containment) and (4) may result in offsite radiation exposures in excess of the limits permitted in normal operation; but the consequences of Class 2 events should not be of such severity as to require interruption or restriction of public use of areas beyond the plant exclusion radius. It should be shown that Class 2 events would not in themselves lead to the occurrence of a Class 3 event.

Class 3 events are accidents of very low probability, postulated in evaluating the design and performance of the plant and the acceptability of the site. In such evaluations the course and consequences of the events are analyzed using very conservative assumptions, and the combination of these unlikely events and the conservative methods of evaluation is characterized by the term "design basis accidents." These postulated accidents will require operation of the engineered safety features and the conservatively calculated potential offsite doses resulting from design basis accidents will be significant, but must be shown to be less than the guideline values given in 10 CFR Part 100.

Table 15-1 lists types of off-design events and accidents that are representative of those that should be evaluated by the applicant in this chapter of the Safety Analysis Report. The applicant should list the off-design events and accidents that have been considered, assign each to the appropriate Class (1, 2, or 3) and justify the Class-assignment by providing the information indicated in the following sections.

15.1 General

This section should provide a brief discussion of the principles and general philosophy upon which the accident analyses are based, and an explanation of any significant differences in approach or scope from that presented in this guide.

15.0 ACCIDENT ANALYSES

The evaluation of the safety of a reactor plant is accomplished, in part, by studies made of the response of the plant to disturbances in process variables and to postulated malfunctions or failures of equipment. Such analyses provide a significant contribution in the selection of the design specifications for components and systems and subsequently serve importantly in showing that a design consistent with public safety has been achieved. These analyses are a focal point of the Commission's construction permit and operating license reviews of reactor facilities.

In previous chapters of the SAR, the individual system and component designs should have been evaluated for effects of anticipated process disturbances and for susceptibility to component malfunction or failures. In this chapter, it is expected that the consequences of those failures or abnormal situations will be examined to evaluate the capability built into the plant to control or accommodate such situations (or to identify the limitations of expected performance).

It is recognized that situations analyzed may range from a fairly common disturbance (such as a loss of electrical load resulting from a line fault) to highly unlikely failures (such as the sudden loss of integrity of a major component). In categorizing the spectrum of situations, for purposes of analysis and presentation in the SAR, the spectrum of abnormal situations, or accidents, generally ranging in severity from minor to very serious, is divided into classes according to radiological consequences as follows:

Class 1 - Events Leading to No Radioactive Release at Exclusion Radius

Class 2 - Events Leading to Small to Moderate Radioactive Release at Exclusion Radius

Class 3 - Design Basis Accidents

Class 1 events are categorized as those which result from any abnormal operational transient and which (1) do not induce fuel failures in excess of those expected during routine operation (i.e., from fuel cladding defects), (2) do not lead to a breach of a barrier to fission product release, or of a primary system boundary, (3) do not require operation of any engineered safety features (although they may require appropriate

TABLE 15-1

REPRESENTATIVE TYPES OF OFF-DESIGN OPERATIONAL TRANSIENTS AND ACCIDENTS TO BE ANALYZED IN CHAPTER 15.0 OF THE SAR

(1) Uncontrolled control rod assembly withdrawal from a subcritical condition, including control rod or temporary control device removal error during refueling.

(2) Uncontrolled control rod assembly withdrawal at power.

(3) Control rod misalignment.

(4) Chemical and volume control system malfunction.

(5) Partial loss of forced reactor coolant flow.

(6) Start-up of an inactive reactor coolant loop or recirculating loop.

(7) Loss of external electrical load and/or turbine, including (for BWRs) closure of main steam isolation valve.

(8) Loss of normal feedwater.

(9) Loss of all AC power to the station auxiliaries (station blackout).

(10) Excessive heat removal due to feedwater system malfunctions.

(11) Excessive load increase, including that resulting from a pressure regulator failure, or inadvertent opening of a relief valve or safety valve.

(12) Anticipated variations in the reactivity load of the reactor, to be compensated by means of action such as buildup and burnout of xenon poisoning, fuel burnup, on-line refueling, fuel followers, temperature, moderator and void coefficients.

TABLE 15-1 (cont'd)

(13) Failure of the regulating instrumentation, causing for example, a power-coolant mismatch. Include reactor coolant flow controller failure resulting in increasing flow.

(14) Possibilities for equipment failures involving loss of component integrity which shifts safety action of instrumentation from one of prevention to one of initiating protective safeguards against the release and dispersal of radioactivity.

(15) External causes such as storms or earthquakes.

(16) Loss of reactor coolant, from small ruptured pipes or from cracks in large pipes, which actuates emergency core cooling.

(17) Minor secondary system pipe break.

(18) Inadvertent loading of a fuel assembly into an improper position.

(19) Complete loss of forced reactor coolant flow.

(20) Waste gas decay tank rupture.

(21) Steam generator tube rupture.

(22) Rod ejection accident (PWR).

(23) Rod drop accident (BWR).

(24) Steamline breaks (BWR).

(25) Steamline breaks (PWRs outside containment).

(26) Break in instrument line or lines from primary system that penetrate containment.

(27) Major rupture of pipes containing reactor coolant up to and including double-ended rupture of the largest pipe in the reactor coolant system (Loss-of-Coolant Accident).

(28) Single reactor coolant pump locked rotor.

(29) Fuel handling accident.

15.2 Class 1 - Events Leading to No Radioactive Release at Exclusion Radius

The evaluation of each Class 1 event or type of event should be presented in a separate sequentially numbered subsection (i.e., 15.2.1 through 15.2.X) containing at least the following information:

15.2.X.1 Identification of Causes - For each situation evaluated there should be included a description of the events that must occur, the order of occurrence and analysis of effects and consequences and the basis upon which credibility or probability of occurrence is determined.

The discussion should show the extent to which reactor protective systems must function, the effect of failure of protective functions, the credit taken for designed-in safety features, reactor protective characteristics, and the performance of backup protective systems, during the entire course of the situation analyzed.

To permit an independent evaluation of the adequacy of the protection instrumentation system as related to safety analyses (e.g., which functions, systems, interlocks, and controls are safety related and what readouts are required by the operator under accident and off-design operational transients) the following should be provided:

(1) Starting conditions and assumptions.

(2) A step-by-step sequence of the course of each accident identifying all protection systems required to function at each step.

(3) Identification of any operator actions necessary.

15.2.X.2 Analysis of Effects and Consequences - The analysis of effects and the attendant consequences should be supported by sufficient information, including, for example:

(1) The methods, assumptions, and conditions, employed in estimating the course of events and the consequences.

(2) The mathematical or physical model employed denoting any simplification or approximations introduced to perform the analyses.

(3) Identification of any digital computer program or analog simulation used in the analysis with principal emphasis upon the input data and the extent or range of variables investigated.

(4) The results and consequences derived from each analysis and the margin of protection provided by either a backup or protective system which is depended upon to limit the extent or magnitude of the consequences.

(5) The considerations of uncertainties in calculational methods, in equipment performance, in instrumentation response characteristics, or other indeterminate effects taken into account in the evaluation of the results.

15.3 Class 2 - Events Leading to Small to Moderate Radioactive Releases at Exclusion Radius

The evaluation of each Class 2 event or type of event should be presented in a separate sequentially numbered subsection (i.e., 15.3.1 through 15.3.X) containing at least the following information:

15.3.X.1 Identification of Causes - The same type and degree of information outlined in Section 15.2.X.1 above should be provided for each Class 2 event.

15.3.X.2 Analysis of Effects and Consequences - The same type and degree of information outlined in Section 15.2.X.2 above should be provided for each Class 2 event, where applicable. In addition, for Class 2 events that may result in an offsite dose, the same type and degree of information outlined in Section 15.4.X.2 should be provided.

15.4 Class 3 - Design Basis Accidents

The evaluation of each Class 3 event or type of event should be presented in a separate sequentially numbered subsection (i.e., 15.4.1 through 15.4.X) containing at least the following information:

15.4.X.1 Identification of Causes - The same type and degree of information outlined in Section 15.2.X.1 should be provided for each of the Class 3 events (design basis accidents) evaluated.

15.4.X.2 Accident Analysis - In addition to the type of information outlined in Section 15.2.X.2 above, the following additional information should be provided:

(1) Explain the conditions and assumptions associated with the accidents analyzed, including any reference to published data or research and development investigations in substantiation of the assumed or calculated conditions.

(2) Identify the time-dependent characteristics, activity, and release rate of the fission products, or other transmissible radioactive materials within the containment system that could escape to the environment via leakages in the containment boundaries and leakage through lines that could exhaust to the environment.

(3) Discuss the extent of systems interdependency (containment system and other engineered safety features) contributing directly or indirectly to controlling or limiting leakages from the containment system, or other sources (e.g., from spent fuel handling areas), such as the contribution of: (a) containment water spray systems, (b) containment air cooling systems, (c) air purification and cleanup systems, (d) reactor core spray or safety injection systems, and (e) post-accident heat removal systems.

(4) Describe the physical or mathematical models used in the analyses and the bases for their use with specific reference to:

(a) the distribution and fractions of fission product inventory assumed to be released from the fuel;

(b) the concentrations of radioactive or fission product inventory airborne in the containment atmosphere and buildup on filters during the post-accident time intervals analyzed.

(c) the conditions of meteorology, topography or other circumstances, and combinations of adverse conditions, considered in the analyses.

(5) Discuss and present the results of calculations of potential integrated whole body and thyroid doses from exposure to radiation as a function of distance and time after the accident. Include specific

results for the two-hour dose at the exclusion boundary and the dose for the course of the accident at the outer boundary of the low population zone for whole body doses from direct radiation, and thyroid doses from inhalation.

(6) Discuss and present the results of calculations of whole body and inhalation doses to personnel in the control room, including the contribution to the doses from personnel ingress and egress to the control room during the course of the accident analyzed. Include the assumptions made with respect to air cleanup systems.

For calculations of loss-of-coolant accidents, use the assumptions given in AEC Safety Guide 3 for Boiling Water Reactors, and Safety Guide 4 for Pressurized Water Reactors. For calculations of steam line break accidents for boiling water reactors, use the assumptions given in Safety Guide 5. For calculations of other design basis accidents use comparably conservative assumptions.

16.0 TECHNICAL SPECIFICATIONS

In accordance with the Atomic Energy Act and Section 50.36 of 10 CFR Part 50, each operating license issued by the Atomic Energy Commission must contain Technical Specifications that include those technical operating limits, conditions, and requirements imposed upon facility operation in the interest of the health and safety of the public. The applicant for an operating license proposes Technical Specifications and bases for his facility which are reviewed by the AEC regulatory staff and modified as necessary before becoming a part of the operating license.

Section 50.36 of 10 CFR Part 50 sets forth definitions and requirements relating to the five categories of Technical Specifications for nuclear reactors, (1) Safety Limits and Limiting Safety System Settings, (2) Limiting Conditions for Operation, (3) Surveillance Requirements, (4) Design Features and (5) Administrative Controls. This section of the regulations also requires that a summary statement of the bases or reasons for such specifications, other than those covering design features and administrative controls, shall be included with each specification, but shall not become part of the Technical Specifications.

Throughout the previous sections of the Standard Format, the necessity for identification of safety limits, limiting conditions and surveillance requirements has been indicated. It is from such information that the Technical Specifications and supporting analyses are developed.

For PSARs

In accordance with Section 50.34 of 10 CFR Part 50, an application for a construction permit is required to include preliminary Technical Specifications. The regulations require an identification and justification for the selection of those variables, conditions, or other items which are determined as a result of the preliminary safety analysis and evaluation to be probable subjects of Technical Specifications for the facility, with special attention given for those items which may significantly influence the final design. The objective of providing preliminary Technical Specifications in the PSAR is to identify those items that would require special attention at the construction permit stage, to preclude the necessity for any significant change in design to support the final Technical Specifications, e.g., particularly those specifications that affect the type,

capacity, or number of components in safety-significant systems. Such components and systems cannot be easily modified after the plant is built and the Final Safety Analysis Report is submitted for approval by the AEC.

The preliminary Technical Specifications and bases proposed by an applicant for his facility should be included in Chapter 16.0 of the Preliminary Safety Analysis Report. The preliminary Technical Specifications should be structured in the same manner as for the proposed Technical Specifications to be provided in the Final Safety Analysis Report. The preliminary Technical Specifications should be complete, i.e., to the fullest extent possible, numerical values and other pertinent data should be provided. For each specification the applicable sections of the PSAR that develop, through analysis and evaluation, the details and bases for the specification should be referenced.

As an alternate to providing complete preliminary Technical Specifications, the applicant may state that his final Technical Specifications will be essentially the same as those for a reference plant of similar design, except for the following two categories of exception:

Category I - Those specifications that do not conform to the applicant's operating practices or his plans for operation of the facility.

Category II - Those specifications that are expected to change as a result of differences between the design of the applicant's plant and that of the reference plant.

If this procedure is followed, for each of the two categories of exceptions defined above, an applicant should provide a list of the specifications for the reference plant that are expected to be different from his plant, and provide alternate specifications appropriate to his plant. Also, for each specification of the reference plant and for each alternate specification for his plant, an applicant should reference the section of his PSAR that develops, through analysis and evaluation, the details and bases for the Technical Specifications.

For FSARs

The Technical Specifications and bases proposed by an applicant for his facility should be included as Chapter 16 of the Final Safety Analysis Report. Except for the specifications covering design features and

administrative controls, each specification selected should be provided with bases in the form of a summary statement of the technical and operational considerations which justify the selection. For each specification the applicable sections of the FSAR which fully develop, through analysis and evaluation, the details and bases for the specification should be referenced.

Additional guidance on the contents of the Technical Specifications, is provided in a document entitled "Guide to Content of Technical Specifications for Nuclear Reactors" prepared by the AEC and available from the Director of Regulation.

17.0 QUALITY ASSURANCE

In order to provide assurance that the design, construction, and operation of the proposed nuclear power plant are in conformance with applicable regulatory requirements and with the design bases specified in the license application, it is necessary that a Quality Assurance Program (QAP) be established by the applicant. In this chapter of the PSAR, the applicant should provide a description of the QAP to be established and executed during the design and construction of the nuclear power plant. In addition, the FSAR should describe the QAP to be established and executed during operation of the nuclear power plant. The QAP must be established at the earliest practical time consistent with the schedule for accomplishing the activity. Where some portions of the QAP have not yet been established at the time the Safety Analysis Report is prepared because the activity will be performed in the future, the description should also provide a schedule for implementation. The program must meet the requirements of Appendix B of 10 CFR Part 50. In order to facilitate the presentation of the information, it is suggested that the QAP for each of the major organizations involved in executing the QAP be described in accordance with the following outline. It is not intended to dictate the format of any QAP Manual; that is left to the descretion of the applicant. It is required, however, that the description address at a minimum each of the criteria in Appendix B in sufficient detail to determine whether all the requirements of the Appendix will be satisfied.

17.1 Quality Assurance During Design and Construction

17.1.1 Organization

Organization charts for the project should be provided that denote the lines and areas of responsibility, authority, and communication within each of the major organizations involved, including those of the applicant, the architect-engineer, the constructor, and construction manager (if different from the constructor). In addition, a single overall organization chart should be included denoting how these companies interrelate for the specific project. These charts and attendent discussions should clearly indicate the organizational location of, organizational freedom of, and authority of the individual or groups assigned the responsibility for checking, auditing, inspecting, or otherwise verifying that an activity has been correctly performed. The charts and discussions should indicate the involvement on the part of the applicant to verify the adequacy of

implementation of the QA programs implemented by the applicant's contractors and suppliers, even for those cases where the applicant has delegated to other organizations the work of establishing and implementing the quality assurance program, or any part thereof.

17.1.2 Quality Assurance Program

The structures, systems, and components to be covered by the QAP should be identified along with the major organizations participating in the program and the designated functions of these organizations. The written policies, procedures, or instructions which implement or will implement the QAP should be described. Where these written policies, procedures, or instructions are not yet effective, a schedule for their implementation should be provided. Sufficient information concerning these written policies, procedures, or instructions should be provided in either this or the following subsections to allow a determination of whether the requirements of Appendix B will be satisfied.

17.1.3 Design Control

A description of the design control measures should be provided. Included should be measures to assure that appropriate quality standards are specified in design documents and that deviations from such standards are controlled; measures for the selection and review of suitability of application of materials, parts, equipment and processes; measures for the identification and control of design interfaces and for coordination among participating organizations; measures for verifying or checking adequacy such as by design reviews, alternate or simplified calculational methods or suitable testing programs; and measures to assure that design changes, including field changes, will be subject to design control measures commensurate with those applied to the original design.

17.1.4 Procurement Document Control

A description of the procurement document control measures should be provided. Included should be measures to assure that applicable regulatory requirements, design bases, and other requirements such as QAP requirements which are necessary to obtain adequate quality are included or referenced in procurement documents.

17.1.5 Instructions, Procedures, and Drawings

A description should be provided of the measures to assure that activities affecting quality will be prescribed by documented instructions, procedures, or drawings and will be accomplished in accordance with these instructions, procedures, or drawings.

17.1.6 Document Control

A description of document control measures should be provided. Included should be measures to assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed.

17.1.7 Control of Purchased Material, Equipment, and Services

A description of the measures used for the control of purchased material, equipment, and services should be provided. Included should be measures for source evaluation and selection; for assessing the adequacy by means of objective evidence of quality furnished by the contractor; for inspection at the contractor source; and for examination of products upon delivery. A description should also be provided of the measures taken to assure that documentary evidence that the material and equipment conform to the procurement requirements is available at the nuclear power plant site prior to installation or use of such material or equipment.

17.1.8 Identification and Control of Materials, Parts, and Components

A description of the measures used for the identification and control of materials, parts, and components should be provided to assure that incorrect or defective items will not be used.

17.1.9 Control of Special Processes

A description of the measures employed for the control of special processes should be provided. Included should be a listing of the special processes and the measures to assure that such special processes are controlled and accomplished by qualified personnel using qualified procedures.

17.1.10 Inspection

A description of the program for the inspection of activities affecting quality should be provided. Included should be an organizational description of the individuals or groups performing inspections, indicating the independence of the inspection group from the group performing the activity being inspected, and a description of how the inspection program for the involved organizations has been or will be established.

17.1.11 Test Control

A description of the test program to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service should be provided. Included should be an outline of the test program; procedures to be developed; means for documenting and evaluating test results of the item tested; and designation of the responsibility for performing the various phases of the program.

17.1.12 Control of Measuring and Test Equipment

A description of the measures used to assure that tools, gages, instruments, and other measuring and testing devices are properly controlled, calibrated and adjusted at specified periods to maintain accuracy within necessary limits should be provided.

17.1.13 Handling, Storage, and Shipping

A description of the measures employed to control handling, storage, shipping, cleaning and preservation of items in accordance with work and inspection instructions to prevent damage or deterioration should be provided.

17.1.14 Inspection, Test, and Operating Status

A description of the measures to indicate the inspection and test status of items to preclude inadvertent bypassing of such inspections and tests should be provided. A description should also be provided of the measures for indicating the operating status of structures, systems, and components of the nuclear power plant to prevent inadvertent operation.

17.1.15 Nonconforming Materials, Parts, or Components

A description of the measures to control nonconforming materials, parts, or components to prevent their inadvertent use or installation should be provided. Included should be the means for identification, documentation, segregation, and disposition of nonconforming material and notification to affected organizations.

17.1.16 Corrective Action

A description of the corrective action measures should be provided to assure that conditions adverse to quality are identified and corrected and that the cause of significant conditions adverse to quality is determined and corrective action taken to preclude repetition.

17.1.17 Quality Assurance Records

A description of the program for the maintenance of records to furnish evidence of activities affecting quality should be provided. Included should be means for identifying the records, retention requirements for the records including duration, location and assigned responsibility, and means for retrieving the records when needed.

17.1.18 Audits

A description of the system of audits to verify compliance with all aspects of the QAP and to determine the effectiveness of the QAP should be provided. Included should be means for documenting responsibilities and procedures for auditing; required frequency of audits; audit results; and designating management levels to which audit results are reported.

17.2 Quality Assurance Program For Station Operation

In the FSAR the applicant should provide a description of the proposed QAP that will govern the quality of all safety related items during operating phase activities. These activities include operating, maintaining, repairing, refueling, and modifying subsequent to the pre-operational phase. The description of the proposed QAP should include each of the QA criteria (Appendix B of 10 CFR Part 50), as outlined in Section 17.1 above.